January 8, 1999 SECY-99-007

FOR: The Commissioners

FROM: William D. Travers /s/
Executive Director for Operations

SUBJECT: RECOMMENDATIONS FOR REACTOR OVERSIGHT PROCESS IMPROVEMENTS

PURPOSE:

This Commission paper provides the staff's recommendations for improving the regulatory oversight processes as requested by the SECY-98-045 Staff Requirements Memorandum (SRM) dated June 30, 1998. This SRM requested that the Commission be informed of the results of the integrated review of the assessment processes (IRAP) public comment period, and requested that the staff forward recommended changes to the assessment process. It was also requested that the staff include any conceptual changes to the inspection program needed to conform with the new assessment process.

This Commission paper also responds to the Commission comments documented in SRM M981102 that resulted from the November 2, 1998, staff briefing on regulatory oversight process improvements. In addition, this paper provides the staff's plans for the continued suspension of the SALP process as requested by the COMSECY-98-024 SRM dated September 15, 1998.

Finally, this paper presents recommendations for improving the NRCs inspection, assessment, and enforcement processes and includes a transition plan for implementing these recommended changes. Although the staff has worked closely with the industry and the public in developing these recommendations, this paper provides the first opportunity to present these recommendations in an integrated manner. The staff requests that the Commission acknowledge that the concepts and scope of the changes presented are consistent with the intent of the referenced SRMs. Recognizing that this proposal is a significant departure from

Contact:
Michael R. Johnson, NRR
301-415-1241

current practice in all areas covered, the acknowledgment would include a positive affirmation on establishing a system of risk informed thresholds and applying them as described; approval of the approaches taken to define information needs, integrate performance indicators with inspection areas and scale regulatory response to findings as illustrated in the assessment matrix. Following the completion of the public comment period the staff will forward the results, along with any necessary changes to the proposals contained herein, for final Commission approval.

SUMMARY

This paper presents recommendations for improving the NRC's reactor oversight processes, including inspection, assessment, and enforcement, and includes a transition plan for implementing these recommended changes. The paper also discusses public comments received on the IRAP proposal and the assessment process in general, and responds to 12 areas of specific interest to the Commission identified in the June 30, 1998 and November 19, 1998 SRMs. The paper informs the Commission of the staff's intention to continue the suspension of the systematic assessment of licensee performance process (SALP) until the new processes have been successfully tried.

The NRC conducted an effort to develop changes to the inspection, assessment, and enforcement processes to improve their objectivity, make them more understandable and predictable, and provide increased focus on aspects of performance that have the greatest impact on safe plant operation. The effort was initiated in response to results of internal reviews and external stakeholder input.

The staff organized three task groups to develop recommended improvements, including a technical framework task group, an inspection task group, and an assessment task group. The activities of each group were closely integrated and all groups interfaced frequently with the public and industry through a series of regularly conducted public meetings in order to provide opportunities to exchange information and receive feedback. The results of these three groups are presented in the attachments to this paper and summarized in the discussion section of the paper. The attachments include the following:

Attachment	Subject
1	Key Figures and Tables
2	Technical Framework
3	Risk-Informed Baseline Inspection Program
4	Assessment Process
5	Enforcement Program Changes
6	Transition Plan
7	Summary of Integrated Review of Assessment Public Comments
8	Commitments (Specific Responses to Staff Requirements Memoranda)

The objective of the technical framework task group was to complete development of a hierarchical regulatory oversight framework; develop performance indicators and appropriate thresholds that could be used to monitor performance within the framework; and

identify aspects of risk-informed inspections that should supplement and verify the validity of the performance indicator data.

The objective of the inspection task group was to develop recommendations for a baseline inspection program that identifies the minimum level of inspection required for a plant (regardless of performance) in order for the NRC to have sufficient information to determine whether plant performance is at an acceptable level. The baseline inspection program was developed by using a risk-informed approach to determine a comprehensive list of areas to inspect (inspectable areas) within the oversight framework. These inspectable areas were selected based on their risk significance. The proposed baseline inspection program is based on several concepts that are fundamentally different than those upon which the current core inspection program is based.

The objective of the assessment task group was to develop a process that will allow the NRC to integrate various information sources relevant to licensee safety performance, make objective conclusions regarding their significance, take actions based on these conclusions in a predictable manner, and effectively communicate these results to the licensees and to the public. The review system developed provides continuous, quarterly, mid-cycle, and end-of-cycle (annual) reviews of licensee performance data (performance indicators and inspection results).

The staff intends to develop changes to the enforcement policy to reflect the recommended changes to the inspection and assessment processes. Although it is too early to propose specific changes, they may include changes in the definitions and thresholds for severity levels to align them with the process and guidance developed for evaluating the safety significance of inspection findings, and changing the criteria for not citing violations to be consistent with the licensee performance results determined by the assessment process.

The staff believes the recommendations that resulted from this effort will address many of the concerns with existing reactor oversight processes. The proposal represents considerable progress, however, continued incremental changes will be necessary to respond to lessons learned during process piloting and implementation. While the recommended process improvements will provide for greater use of objective information and defined thresholds for regulatory action, the proposed process still includes some level of judgement, especially in the application of a graded regulatory response to declining licensee performance. The process is intended to provide minimal regulatory interaction beyond the baseline inspection for good performers and a strong regulatory response for facilities that approach unacceptable performance. Finally, although these improvements decrease the reliance on subjective decisions, some level of judgement will still be required because of the complexity of nuclear plant activities and the variability between plants.

The staff is asking the Commission to approve the scope and concepts of the recommended changes to the regulatory oversight processes, and their continued development and implementation as described in the attached transition plan.

<u>BACKGROUND</u>:

On March 9, 1998, the staff issued SECY-98-045, “Status of the Integrated Review of the NRC Assessment Process for Operating Commercial Nuclear Reactors,” which forwarded the staff’s recommendation for a new integrated assessment process. The fundamental concepts that formed the basis of the IRAP proposal were: (1) inspection findings provided the basis for the assessment, (2) inspection findings would be categorized by performance template areas and would be scored according to safety significance, (3) assessment would be accomplished by totaling the scores in each template area and comparing these scores against threshold values, and (4) NRC actions would be taken based on a decision model. On April 2, 1998, the staff briefed the Commission on the staff proposal described in SECY-98-045.

On June 30, 1998, the Commission issued the SRM for SECY-98-045, in which the Commission expressed concerns with (1) the apparent use of enforcement as a “driving force” for the assessment process, (2) the quantitative scoring of plant issues matrix (PIM) entries, and (3) the use of color coding to define performance rating categories. However, the Commission did approve the solicitation of public comment on the IRAP proposal, and requested the staff to (1) provide a recommendation for changes to the assessment process, (2) address regional consistency and equitable treatment of plants receiving varying levels of inspection effort, and (3) include conceptual changes to the inspection program needed to conform with the new assessment process.

In parallel with the staff’s development of the IRAP proposal, the industry developed an independent proposal for improving the assessment process. This effort, led and coordinated by the Nuclear Energy Institute (NEI), resulted in a concept that was fundamentally and philosophically different from the IRAP proposal. This approach established tiers of licensee performance based on maintaining the barriers to radionuclide release, minimizing events that could challenge the barriers, and ensuring that systems can perform their intended functions. Performance in these tiers would be measured through reliance on high-level, objective indicators with thresholds set for each indicator to form a utility response band, a regulator response band, and a band of unacceptable performance.

In response to the NEI proposal, Commission comment on the IRAP proposal, and comments made at the July 17, 1998, Commission meeting with public and industry stakeholders and the July 31, 1998, hearing before the Senate, the staff set out to develop a single set of recommendations for making improvements to the regulatory oversight processes.

The IRAP public comment period and a series of public meetings were used to facilitate internal and external input into the development of these recommendations. The 60-day IRAP public comment period, which ended on October 6, 1998, was used to seek comment on improvements to the assessment process. As part of the public comment period, the staff sponsored a 4-day public workshop from September 28 through October 1, 1998, to interact with the industry and public to obtain and evaluate input on improving the regulatory oversight processes. During the workshop a consensus was reached on the overall philosophy for regulatory oversight and general agreement was achieved among workshop participants on the defining principles for the oversight processes.

After the workshop, the staff began several short-term activities to continue developing the improvements to the regulatory oversight process that had been initiated at the workshop. All of these activities were coordinated and integrated and involved broad participation from all four regions, the Office of Nuclear Reactor Regulation (NRR), the Office of Enforcement (OE) , the Office of Nuclear Regulatory Research (RES), and the Office for Analysis and Evaluation of Operational Data (AEOD). The staff selected to participate in these activities were agency experts in various aspects of regulatory oversight, such as risk analysis, use of performance indicators, inspection, and assessment techniques. Each of these activities also involved frequent interaction with the industry and the public during the development of recommended improvements.

Three task groups were formed to develop these recommendations: a technical framework task group, an inspection task group, and an assessment task group. The technical framework task group was responsible for completing the regulatory oversight structure and for identifying the performance indicators (PIs) and appropriate thresholds that could be used to measure performance. The inspection task group was responsible for developing the scope, depth, and frequency of a risk-informed baseline inspection program that would be used to supplement and verify the PIs. The assessment process task group developed methods for integrating PI and inspection data, determining NRC action based on assessment results, and communicating results to licensees and the public. OE activities to improve the enforcement process were coordinated with these three task groups to ensure that enforcement process changes were properly evaluated in the framework structure, and that changes to the inspection and assessment programs were integrated with changes to the enforcement program.

The staff briefed the Advisory Committee on Reactor Safeguards (ACRS) on the results of the workshop and the progress of these activities on October 2, November 20, and December 3, 1998. The staff briefed the Commission on the progress of these efforts on November 2, 1998. On November 19, 1998, SRM M981102 was issued in response to this Commission briefing and directed that the staff (1) refine key definitions, (2) identify attributes that are important to the assessment program but are not covered by performance indicators, (3) identify the types of information and methodology used in an assessment process, (4) identify the desired outcomes of the cornerstones, (5) further identify the proposed vehicles to inform the Commission and public of the assessment results, and (6) provide the methodology the staff will use to verify and validate the efficacy of the improved oversight process.

The following discussion details the need for change to the regulatory oversight processes, the approach taken by the task groups to develop recommendations for process improvements, and the results of the work accomplished by these task groups.

DISCUSSION

Regulatory Principles and The Need for Change

Several important principles form the basis for how the NRC oversees and regulates licensed activities. As stated in the Atomic Energy Act of 1954, as amended, one of the missions of the NRC is to ensure that commercial nuclear power plants are operated in a manner that provides adequate protection of public health and safety and the environment and protects against radiological sabotage and the theft or diversion of special nuclear materials.

Through the NRC's Principles of Good Regulation (Independence, Openness, Efficiency, Clarity, and Reliability), the NRC can instill confidence in the public that these facilities are regulated in a manner that meets this mission. An independent regulatory oversight process is one in which the agency's decisions are based on unbiased assessments of licensee performance. An open oversight process provides an opportunity for public awareness of process results. An efficient oversight process is one that applies agency resources in a risk-informed manner. A clear oversight process will result in agency actions that are logical and coherent, with a nexus to agency regulations and goals. And a reliable oversight process will result in agency actions that are predictable, transparent, and that have a clear tie to regulations.

Commercial nuclear power plants have been operated safely with overall plant performance, as indicated by trends in both NRC and industry performance indicators, improved over the last 10 years. This improvement in plant performance can be attributed, in part, to successful regulatory oversight in accordance with these principles. Despite this success, the agency has noted that the current inspection, assessment, and enforcement processes (1) are at times not clearly focused on the most safety important issues, (2) consist of redundant actions and outputs, and (3) are overly subjective with NRC action taken in a manner that is at times neither scrutable nor predictable.

These concerns and observations have been recently echoed by external stakeholders such as the Congress, the industry, and the public. In light of these noted weaknesses and stakeholder feedback, the Commission has identified the opportunity to improve the regulatory oversight of licensees, and has directed the staff to develop improvements to these processes. The overall objective of developing improvements to these processes was to:

- Improve the objectivity of the oversight processes so that subjective decisions and judgment were not central process features.

- Improve the scrutability of these processes so that NRC actions have a clear tie to licensee performance.

- Risk-inform the processes so that NRC and licensee resources are focused on those aspects of performance having the greatest impact on safe plant operation.

The recommendations made in this paper are intended to improve public confidence in the oversight of licensed activities, and increase the effectiveness and efficiency of the NRC, while ensuring that the agency's mission to protect public health and safety is still met.

<u>Objectives and Approach</u>

The staff used a top-down, hierarchical approach to develop the concept for a new regulatory oversight framework that implements this change vision and addresses the agency's regulatory principles. This approach starts with a desired outcome, identifies performance goals to achieve this outcome, and then identifies specific objectives and information needs to meet each performance goal. The regulatory oversight framework developed by the staff using this approach is represented in Attachment 1, Figure 1. This framework starts at the highest level, with the NRC's overall mission to ensure that commercial nuclear power plants are operated in a manner that provides adequate protection of public health and safety.

The staff then identified those aspects of licensee performance that are important to the mission and therefore merit regulatory oversight. The NRC Strategic Plan identifies the performance goals to be met for ensuring nuclear reactor safety and include the following:

- Maintain a low frequency of events that could lead to a nuclear reactor accident;

- Zero significant radiation exposures resulting from civilian nuclear reactors;

- No increase in the number of offsite releases of radioactive material from civilian nuclear reactors that exceed 10 CFR Part 20 limits; and

- No substantiated breakdown of physical protection that significantly weakens protection against radiological sabotage, or theft or diversion of special nuclear materials.

These performance goals reflect those areas of licensee performance for which the NRC has regulatory responsibility in support of the overall agency mission. These performance goals were represented in the framework structure as the strategic performance areas of Reactor Safety, Radiation Safety, and Safeguards, and formed the second level of the regulatory oversight framework.

With a risk-informed perspective, the staff then identified the most important elements in each of these strategic performance areas that form the foundation for meeting the overall agency mission. These elements were identified as the cornerstones in the third level of the regulatory oversight framework structure. These cornerstones serve as the fundamental building blocks for the regulatory oversight process, and acceptable licensee performance in these cornerstones should provide reasonable assurance that the overall mission of adequate protection of public health and safety is met.

Once the regulatory oversight framework was established, the staff developed defining principles that formed the strategy and rules for the further development of the details of the regulatory oversight processes. These defining principles were developed with internal and external input obtained through written comments and public meetings such as the 4-day workshop. These defining principles established the relationship between elements of the oversight processes, such as enforcement and inspection.

- There will be a risk-informed baseline inspection program that establishes the minimum regulatory interaction for all licensees.

- Thresholds can be set for licensee safety performance, below which increased NRC interaction (including enforcement) would be warranted.

- Adequate assurance of licensee performance at the cornerstone level requires assessment of both PIs and inspection findings.

- Both the PIs and results of inspections used to assess a cornerstone will have risk-informed thresholds.

- Crossing a PI threshold and an inspection threshold will have the same meaning with respect to safety significance and the need for some level of NRC interaction.

- The baseline inspection program will cover those risk-significant attributes of licensee performance not adequately covered by PIs.

- The baseline inspection program will also verify the accuracy of the PIs and provide for event response.

- Enforcement actions taken (e.g., the number of cited violations, the amount of a civil penalty) should not be an input into the assessment process. However, the issue that led to the enforcement action will continue to be considered in the assessment.

- Assessment process results might be used to modulate enforcement actions (although assessment results would not affect the determination of violation severity level).

- Guidelines will establish criteria for identifying and responding to unacceptable licensee performance.

It is important to note that these defining principles will result in an oversight process that provides adequate margin in the assessment of licensee performance so that appropriate licensee and NRC actions are taken before unacceptable performance occurs.

Summary of Task Group Activities and Results

Once the framework structure and defining principles were established, the staff then had the basis for determining what information was needed to provide reasonable assurance that the agency's mission was being achieved. As previously discussed, task groups were formed to finalize the regulatory oversight framework structure, develop a new baseline inspection program, develop a new assessment process, and coordinate with enforcement process improvements. The following sections provide a summary of the activities of these task groups and the results of their work. Those key figures and tables referenced in the following discussion are included as Attachment 1 to this paper.

Regulatory Oversight Framework

The goals and objectives of the technical framework task group's activities were to identify and develop:

- the cornerstones of safety and the key attributes of performance within each cornerstone;

- the performance indicators that can be used to assess performance in certain areas;

- performance indicator thresholds intended to establish clear demarcation points for identifying fully acceptable, declining, and unacceptable levels of performance;

- aspects of risk-informed inspections that should supplement and verify the validity of the performance indicator data.

The task group also evaluated cross-cutting issues, benchmarked the proposed performance indicators against prior plant performance, and identified future development activities. During this effort, information was shared with the inspection and assessment process task groups for use in developing a new baseline inspection program and overall NRC reactor assessment process. Details of the results of the technical framework task group's efforts are included as Attachment 2 to this paper.

As a starting point, the technical framework task group used the results of the Performance Assessment Public Workshop held from September 28 through October 1, 1998. During this workshop, general agreement was reached with the industry and members of the public on the regulatory oversight framework and the cornerstones of safety.   A diagram of this framework showing the relationship between the NRC's overall safety mission, strategic performance areas, and cornerstones of safety is provided in Attachment 1, Figure 1.

These cornerstones of safety were chosen to (1) limit the frequency of initiating events; (2) ensure the availability, reliability, and capability of mitigating systems; (3) ensure the integrity of the fuel cladding, reactor coolant system, and containment boundaries; (4) ensure the adequacy of the emergency preparedness functions; (5) protect the public from exposure to radioactive material releases; (6) protect nuclear plant workers from exposure to radiation; and (7) provide assurance that the physical protection system can protect against the design-basis threat of radiological sabotage.

Within each cornerstone area, the task group then used a top-down, hierarchical, risk-informed approach to:

- identify the objective and scope of the cornerstone;
- identify the desired results and important attributes of the cornerstone;
- identify what should be measured to ensure that the cornerstone objectives are met;
- determine which of the areas to be measured can be monitored adequately by performance indicators
- determine whether inspection or other information sources are needed to supplement the performance indicators, and
- determine the thresholds of performance for each cornerstone, below which additional NRC actions would be taken.

Where possible, the task group sought to identify performance indicators as a means of measuring the performance of key attributes in each of the cornerstone areas. Where such a performance indicator could not be identified, the group proposed a "complementary" inspection activity. Where a performance indicator was identified but was not sufficiently comprehensive, the group proposed "supplementary" inspection activities. The task group also identified the need for "verification" type inspections to verify the accuracy and completeness of the reported performance indicator data. These recommended inspection activities were provided to the risk-informed baseline inspection task group for consideration in developing the baseline inspection program.

Performance indicators, together with risk-informed baseline inspections, are intended to provide a broad sample of data to assess licensee performance in the risk-significant areas of each cornerstone. They are not intended to provide complete coverage of every aspect of plant design and operation. It is recognized that licensees have the primary responsibility for ensuring the safety of the facility. Objective performance evaluation thresholds are intended to help determine the level of regulatory engagement appropriate to licensee performance in each cornerstone area. Furthermore, based on past experience it is expected that a limited number of risk-significant events will continue to occur with little or no indication of declining performance. Follow up inspections will be conducted to ensure that the cause of these events are well understood and that licensee corrective actions are adequate to prevent recurrence. Likewise, reactive inspections may be performed to follow up on allegations. The results of these follow up inspections will be factored into the assessment process along with performance indicators and risk-informed baseline inspections.

The performance indicators selected for each cornerstone, along with performance thresholds, are listed in Attachment 1, Table 1 to this paper. These thresholds were selected for consistency with the performance threshold conceptual model provided in Attachment 1, Table 2. They correspond to levels of performance requiring no additional regulatory oversight (above the green-to-white threshold), performance that may result in increased oversight (below the green-to-white threshold), performance that will result in specific NRC actions (below the white-to-yellow threshold), and performance that is unacceptable (below the yellow-to-red threshold). For some PIs, white-to-yellow or yellow-to-red thresholds were not identified, because the indicators could not be directly tied to risk data. As experience is obtained, and additional PIs become available, the PIs and thresholds are likely to be refined. It should be noted that although not expected, should a licensee's performance reach what has been determined to be an unacceptable level, margin would still exist before an undue risk to public health and safety would be presented. As later described in the assessment process section of this paper, the extent of NRC actions would be graded based upon the relative deviation from the performance indicator threshold and the number of thresholds exceeded.

Once the performance indicators and corresponding thresholds were selected, the task group performed a benchmarking analysis to compare the indicators against several plants that had been previously designated by the agency as having either poor, declining, average, or superior performance. The analysis indicated that the performance indicators could generally differentiate between poor and superior plants, but were not as effective at differentiating average levels of performance. In some instances, the cause of the poorly rated plants was due to design or other issues for which valid performance indicators have not been developed. Issues such as these are within the scope of the risk-informed baseline inspection program.

The task group also identified aspects of licensee performance (such as human performance, the establishment of a safety conscious work environment, common cause failure, and the effectiveness of licensee problem identification and corrective action programs) that are not identified as specific cornerstones, but are important to meeting the safety mission. The task group concluded that these items generally manifest themselves as the root causes of performance problems. Adequate licensee performance in these crosscutting areas will be inferred through cornerstone performance results from both PIs and inspection findings.

Risk-Informed Baseline Inspection Program

The objective of the inspection task group was to develop recommendations for a baseline inspection program that is risk-informed and that identifies the minimum level of inspection required for a plant (regardless of performance) in order for the NRC to have sufficient information to determine whether plant performance is at an acceptable level. A key input to the group's project was the regulatory oversight framework, developed by the technical framework task group. The inspection task group accomplished this objective, and the recommended program is described in Attachment 3 to this paper.

The baseline inspection program was developed by using a risk-informed approach to determine a comprehensive list of areas to inspect (inspectable areas) within each cornerstone of safety. These inspectable areas were selected based on their risk significance (i.e., they are needed to meet a cornerstone objective as derived from a combination of probabilistic risk analyses insights, operational experience, deterministic analyses insights, and requirements in regulations). The final list of inspectable areas incorporated those inspection areas recommended by the technical framework task group and is presented in Attachment 1, Table 3.

The scope of inspection within each inspectable area was determined using the same risk-informed approach. The scope of inspection was also modified by the applicability of a performance indicator. The more fully an indicator measures an area, the less extensive is the scope of inspection.

Several documents were created to integrate risk insights into the baseline inspection program and to aid inspectors and regional managers. Basis documents were created to describe the scope of each inspectable area and the justification for inspection based on risk information. The basis documents also were used to indicate whether the inspection is designed to be complementary or supplementary to a performance indicator (Part 1 of the program) or designed only for verification of a performance indicator (Part 2 of the program). Risk information matrices (RIMs) were developed with input from the Office of Nuclear Regulatory Research to serve as guides in planning and conducting inspections as described in Attachment 3, Section 1.3. Data sources for these RIMs are referenced at the end of RIM No. 1 in Attachment 3.

Inspection practices at two Federal Government agencies were reviewed to determine how they used risk insights to establish the level of inspection effort. The staff held discussions with the Safety, Health, and Environmental Management Division of the Environmental Protection Agency, and reviewed a recent General Accounting Office report, GAO/RCED-98-6, "Weaknesses in Inspection and Enforcement Limit FAA [Federal Aviation Administration] in Identifying and Responding to Risks." The number of inspections and the allotted resources varied widely. Neither of these agencies used probabilistic risk assessment techniques to establish inspection areas or effort. In general, these organizations based their inspections upon regulatory requirements, failure history of the item being inspected, and judgement. The lessons learned by these agencies were: (1) inspections provide both an indirect measure of the industry's compliance and an early warning of potential safety and security problems, (2) more intensive (but less frequent), independent, structured team inspections are more effective than routine inspections performed by individual inspectors, (3) inspection protocols (checklists or other job aids based on safety-critical elements) provide more systematic, comprehensive, and consistent inspections, and (4) inspection findings that have generic applicability should be fed back to the industry. The insights gained from these agencies will be used in developing the more detailed guidance documents for the baseline inspection program.

The recommended baseline program contains certain concepts that are a change in the approach to conducting an inspection program from that currently used in Inspection Manual Chapter (IMC) 2515. The key concepts are summarized below:

- The program is the minimum level of inspection conducted at all power reactor facilities, regardless of their performance. Licensees performing at a level not requiring additional NRC interaction will only be inspected at the baseline inspection level of effort.

- Increases above the baseline program will be termed reactive and initiative inspections as in the current IMC 2515. This increased inspection effort will be based on criteria specified in the assessment process to address declining licensee performance, or in response to an event, and is not included in the baseline program.

- The scope of the baseline program is defined by inspectable areas linked to the cornerstones of safety. The justification for inclusion of the inspectable area in the baseline program is described in a basis document.

- The baseline program has three parts: (1) inspection in inspectable areas in which PIs are not identified and in which PIs do not fully cover the inspectable area; (2) ongoing verification of the information provided in performance indicators; and (3) comprehensive review of licensee effectiveness in identifying and resolving problems.

- The process for planning inspections will be based on a 12-month cycle, aligned with the NRC's fiscal year. The planning process will be guided by the RIMs and with plant-specific data. Information in the RIMs can be modified to reflect site-specific risk insights.

- Budgeted inspection resources are based on insights specified in the RIMs. These resources are fixed within a cornerstone of safety, but may be shifted between inspectable areas within a cornerstone as plant activities dictate.

- Procedures will guide inspectors through their review of licensee activities. The procedures will be a brief checklist of key methods to use during review of each inspectable area in a cornerstone.

Many details of the recommended program were developed by the inspection task group, but more work needs to be completed before implementing such a program. This work has been incorporated into the transition plan, which is discussed later in this paper.

Assessment Process

The charter of the assessment task group was to develop a process that will allow the NRC to integrate various information sources relevant to licensee safety performance, make objective conclusions regarding their significance, take actions based on these conclusions in a predictable manner, and effectively communicate these results to the licensees and to the public. This effort focused on the design of an assessment process within the regulatory oversight structure and was closely coordinated with the framework, inspection, and enforcement efforts. The details of the recommended changes to the assessment process are given as Attachment 4 to this paper.

The following key principles were identified as having a direct effect on the assessment process design:

- Both performance indicators (PIs) and inspection results will be inputs to the assessment process.

- Performance indicators and cornerstone inspection areas (inspection results grouped by cornerstone area) will have established thresholds.

- Crossing PI or cornerstone inspection area thresholds will have similar meaning and will result in the NRC considering a similar range of actions.

A review system, shown in Attachment 1, Table 4, was developed that provides continuous, quarterly, mid-cycle, and end-of-cycle (annual) reviews of licensee performance data (PIs and inspection results). The system is designed so that the lower level reviews are informal reviews of performance data and are not resource intensive. The mid-cycle review is more formal and is focused on assessing performance to determine appropriate NRC inspection actions. The mid-cycle review generates an inspection planning letter. The end-of-cycle review generates both an assessment report and an inspection planning letter. The agency action review is reserved for plants requiring consideration of agency-wide actions. This review is analogous to the review performed at the current senior management meeting (SMM), however the focus has been changed from an assessment activity to an oversight and agency-level action approval function.

An action matrix, shown in Attachment 1, Table 5, was developed to provide guidance for consistent consideration of actions. The actions are graded across five ranges of licensee performance in all response categories (management meeting, licensee action, NRC inspection, and regulatory actions) and in terms of annual communication of assessment results. Action decisions are triggered directly from the threshold assessments of PIs and cornerstone inspection areas. For example, a single PI or cornerstone inspection area crossing its threshold would require the NRC to <u>consider</u> the actions listed in the second performance range of the action matrix, such as regional initiative inspection to determine the cause of the assessment input degradation. More significant changes in performance, such as one degraded cornerstone, would lead to the consideration of more significant actions.

The action matrix is not intended to provide guidance that is excessively rigid. It establishes expectations for interactions, licensee actions, and NRC actions. It does not preclude taking less action or additional action, when justified. The key point is that assessment results are not altered; action decisions are modified, when appropriate.

The communication of assessment results involves quarterly updates of assessment data, semiannual inspection planning letters, and annual assessment reports. All assessment results and NRC actions will be forwarded to Commission via a negative consent Commission paper before an annual Commission meeting. <u>All</u> assessment results are released at the Commission meeting to provide proper balance and context. This differs from the current SMM, which focuses primarily on poor performers.

Enforcement Process

The staff intends to develop changes to the enforcement policy for power reactors to reflect the recommended changes to the inspection and assessment processes. The fundamental purposes of the NRC enforcement policy need not be changed. However, changes in the definitions and thresholds for severity levels will need to be aligned with the process and guidance developed for evaluating the safety significance of inspection findings. Additionally, the criteria for not citing violations should be tied to the licensee performance results determined by the assessment process. For example, the NRC may request a licensee to document corrective actions for current and previous related deficiencies when licensee performance degrades into the increased regulatory response performance band. Additionally, for those plants in the utility response band, the NRC would not combine violations of low safety significance into an escalated enforcement action. Attachment 5 discusses some preliminary views on how the enforcement policy and program might be changed. However, it is premature to develop specific changes until the oversight processes are more fully developed.

Conclusion

The staff achieved its objective of developing improvements to the regulatory oversight process that address each of the needs for change discussed earlier in this Commission paper: increase objectivity, improve scrutability, reduce redundancy, and risk-inform the process.

The proposed process will provide for increased objectivity by relying on objective performance indicators, where possible, to provide the basis for determining performance, and using risk-informed thresholds to determine expected regulatory and licensee response.

The proposed process is more scrutable by more clearly relating individual information from inspections and performance indicators to their impact on overall safety performance. This will serve to produce a clearer trail of evidence and uses the action matrix to trigger NRC actions in a logical and consistent manner, with a clear tie to licensee performance.

The proposed process has eliminated many of the redundancies of the current processes by developing an single, integrated assessment process that sends a clear message regarding licensee performance. The assessment and enforcement processes are also more closely aligned and integrated to prevent redundant and conflicting messages on licensee performance.

The new process is designed to be risk-informed. The risk significance of performance data is the primary determinant of data significance in the process, particularly in the new risk-informed baseline inspection program. PI and cornerstone inspection area thresholds include risk insights, where applicable.

The staff recognizes the need to accommodate future changes to these processes in response to issues such as the identification of new, risk-significant generic safety issues and lessons learned from implementation. While the recommended process improvements described in this paper will provide a better framework for oversight, assessments of licensee performance will continue to be only as good as the performance data and inspection findings that feed it. Further, while these improvements decrease the reliance on subjective decisions, some level of judgement will still be required due to the complexity and the variability between plants.

Several key policy issues remain that must be considered in arriving at a final process for implementation. In addition, although significant progress was made in developing concepts for the future regulatory oversight process, much work remains in benchmarking, piloting, developing implementation procedures, and training on the new process. The key policy issues are:

- Evaluating the interface with 10 CFR Part 50. The new oversight process increases focus on certain risk-significant requirements and decreases focus on certain other requirements. This could result in situations where low significance findings, even if numerous, would be evaluated and treated as such.

- Revisiting event response and evaluation processes. The new process recognizes that a certain number of random, significant events are possible (industry wide) without necessarily having an impact on assessment conclusions. That is because the process would evaluate the event within the context of overall performance.

- Revisiting the n+1 policy for resident inspector staffing. The proposed oversight process recommends that only a baseline inspection level of effort be performed at certain plants. This may conflict with the n+1 policy.

- Organizational impact. Regional and headquarters organizational structures may need to be changed to support the framework and oversight processes.

A transition plan and success criteria have been drafted to guide future development efforts.

<u>Transition Plan</u>

The staff has developed a recommended plan to be used by the NRC to transition through the implementation of the revised oversight process. This transition plan includes change management strategies for creation of management systems necessary to support those desired changes. These aspects are key ingredients in enabling an organization to successfully implement change. The details of this transition plan appear in Attachment 6 to this paper.

The transition plan contains milestones for both the NRC and industry. Successful implementation will require a continuing interface with the industry and other stakeholders at various stages. Significant investment in staff and management resources also will be required to complete the necessary supporting documents and infrastructure, develop and train staff, and manage all aspects of the resulting change effort.

The transition plan contains challenging but achievable goals. The milestones reflect best estimates based on recognized challenges. Adjustments will be made as necessary to allow for resolution of unanticipated problems (e.g., difficulty in assigning significance to inspection findings, difficulty in collecting PI data in a consistent manner, unexpected change in resources) or additional direction from the Commission.

A key factor during the implementation of the new process focuses on creating and maintaining a shared vision within the NRC. “Opinion Leaders” are individuals within the organization who have significant credibility among their peers so that their peers’ views are influenced by the opinion leader’s views. The identification and cultivation of opinion leaders at both the regional and Headquarters offices will be important for creating alignment within the agency and extending that vision to other stakeholders. These opinion leaders will be the “agents of change” within the NRC and will form the “Change Coalition.” The Change Coalition will be the communication ambassadors at all levels within the agency. This group will discuss the need for change, what the changes will be, and how the change will be accomplished. It is anticipated that the industry will be conducting a similar process during program implementation.

A Transition Task Force, which is separate from the Change Coalition, will be formed in order to manage the phase-out of the existing processes and the phase-in of the new oversight processes. The role of the Transition Task Force will be to complete the development of the detailed implementing instruments and infrastructure.

A major feature of the transition plan will include piloting the process at two sites in each region for six months. The results of the pilot program will be measured against previously established success criteria prior to proceeding with full implementation. Training will be provided to the staff throughout the process culminating in a joint NRC/stakeholder workshop prior to full implementation. Existing processes such as plant performance reviews (PPRs) and SMMs will be phased out as they are replaced by the new risk-informed oversight process.

The pilot program is just one aspect of a multi-pronged approach that will be used for measuring the success of regulatory oversight process improvements. In addition to the pilot program, PI and inspection finding significance benchmarking will be performed for a limited number of plants to determine the technical feasibility of the new process. Further, the overall oversight process will be evaluated after about one year of full implementation. This evaluation will verify that the oversight process objectives are being met. Potential success criteria are shown in Table 4.1 of Attachment 4 to this Commission paper.

Public Comment

As directed by the June 30, 1998, SRM for SECY-98-045, public comment was solicited on the IRAP proposal and the assessment process in general. The *Federal Register* notice that announced the 60-day public comment included a questionnaire to focus public comment on specific topics. This questionnaire grouped these topics into four broad categories; Regulatory Oversight Approach, Integrated Assessment Process, Risk-Informed Assessment Guidance, and Indicators.

There were 26 respondents to the *Federal Register* notice. Industry groups, represented by NEI, licensees, support contractors, and law firms, submitted 19 of the responses. Public advocacy groups submitted 3 of the responses, concerned citizens or consultants submitted 3 responses, and one of the responses came from a State government. A summary and evaluation of these comments can be found in Attachment 7 to this paper. These public comments were evaluated and considered during the development of the regulatory oversight improvements described herein. Public input was appropriately reflected in the recommended changes to the inspection, assessment, and enforcement processes.

Regarding the regulatory oversight approach, industry groups stated that thresholds for NRC action should be based on objective and measurable performance indicators that relate to protecting public health and safety. These thresholds should be a blend of regulatory requirements and risk insights. One member of the public responded that the NRC must establish a threshold at which underperforming plants must be shutdown. The majority of respondents supported the use of performance indicators and stated that the use of PIs would be timely and comprehensive enough to ensure the adequate protection of public health and safety. The majority of respondents supported the enhanced use of licensee self-assessments and felt that this would result in a regulatory process that was sufficiently independent.

For an integrated assessment process, all respondents agreed that the NRC should not formally recognize superior performing plants, and the majority of respondents did not support the continuation of the watch list. The vast majority of respondents stated that positive inspection findings should not be factored into the assessment process. Industry groups supported an approach similar to the NEI proposal as a means to provide a quantitative input into the assessment process. There were a wide variety of responses to the periodicity of assessment with some respondents supporting an annual assessment and other respondents making alternate proposals.

Several respondents stated that risk insights can be used to identify risk-important plant indicators and to set thresholds for performance. Further, the comments indicated that the guidance in Regulatory Guide 1.174 can be used to establish safety thresholds for the performance of risk-significant structures, systems, and components. Several respondents stated that issues involving human performance and risk management that affect safety performance will be reflected in the performance indicators. If poor human performance or other causes result in performance falling below PI thresholds, then the NRC should initiate action to address these issues.

Regarding the use of indicators, industry groups stated that the NRC should base its assessment on objective indicators with risk-informed thresholds to directly measure safety performance. Respondents stated that the indicators and thresholds proposed by NEI provide a more direct indicator of safety and trends in performance than the current NRC indicators and trending methodology. However, one member of the public noted that longstanding design problems are not accurately reflected in safety system reliability variables, and inconsistent reporting by licensees results in the licensee event report (LER) database not being an accurate source of data on nuclear plant problems. The majority of respondents also stated that financial indicators should not be used in the assessment process. Licensee financial information is an issue for utility management and the financial community, and financial indicators are not a predictor of safety outcomes or plant safety.

SALP Suspension

The SRM for COMSECY-98-024, dated September 15, 1998, approved the staff's recommendation for suspending the SALP process and directed the staff to inform the Commission of its plans relative to whether the SALP process should be resumed in the future or terminated.

The staff intends to continue with the suspension of the SALP process and continue with the current assessment processes, including an annual senior management meeting. As described in Attachment 4, the recommended changes to the assessment process will not require the performance of SALP assessments. Therefore, in accordance with the transition plan as described in Attachment 6, the staff will continue the suspension of SALP until the pilot program for the recommended process improvements is completed. Assuming the pilot program is successful, the staff will propose to delete the SALP program and cancel the associated program documents.

RESOURCES

Considerable resources will be required in the short term to implement these changes. As described in the attached transition plan, the staff initially estimates that approximately 17-19 FTE will be required to develop and implement the recommended changes, including training. This is in addition to the 6.5 FTE expended to date in FY 1999 for the development of these recommendations. These FTEs are within the currently budgeted resources in FY 1999 and FY 2000 for developing and implementing changes to the inspection and assessment programs. These activities have been included in the Reactor Performance Assessment Program and Inspection Program operating plans .

In the long term, the recommended changes to the regulatory oversight processes described herein will likely result in overall reductions in the resources required for program implementation. For example, inspection program changes recommended by the risk-informed baseline inspection program will likely result in fewer hours of direct inspection effort per power reactor unit than is currently allotted in the core inspection program. Further, changes in the scope, depth, and frequency of the baseline inspection program as compared to the current core program will likely result in changes in the division of responsibility between region-based and resident inspectors. Changes to the assessment process are likely to result in fewer resources required to assess licensee performance, decide on appropriate NRC action, and communicate these assessment results to the licensees and the public. Changes to the enforcement policy will likely result in fewer resources required to document and follow up on regulatory discrepancies with no safety significance.

Although overall resource savings are expected in the long term, it would be premature to make any resource reduction decisions at this time beyond those already documented in the FY 2000 budget submittal. The staff will be able to further quantify these resource changes once procedure development is complete and the process is implemented at the pilot plants.

COMMITMENTS

The SECY-98-045 SRM dated June 30, 1998, and SRM M981102, issued in response to the November 2, 1998, Commission briefing on reactor oversight process improvements identified 12 specific areas of Commission interest. The areas and how they are addressed in this Commission paper are summarized in Attachment 8.

COORDINATION:

The Office of the General Counsel has reviewed this Commission paper and has no legal objections to its content.

The Office of the Chief Information Officer has reviewed this Commission paper for information technology and information management implications and has no objections.

The Office of the Chief Financial Officer has reviewed this Commission paper for resource implications and has no objections.

RECOMMENDATIONS: That the Commission:

1. Acknowledge that the concepts and scope of the changes presented are consistent with the intent of the referenced SRMs. This would include a positive affirmation on establishing a system of risk informed thresholds and applying them as described; approval of the approaches taken to define information needs, integrate performance indicators with inspection areas and scale regulatory response to findings as illustrated in the assessment matrix. Final approval would be sought following the comment period in March.

2. Note:

   a. Unless directed otherwise, the staff will continue with development efforts (e.g., stakeholder meetings and procedure development) as outlined in the attached transition plan,

   b. The proposed schedule for transition to the new processes (Attachment 6), is contingent upon the staff receiving a response from the Commission by March 31, 1999.

   c. The request for comment on the process recommendations described herein will be published in the *Federal Register* for a 30-day public comment period.

William D. Travers
Executive Director
for Operations

Attachments:
1. Key Figures and Tables
2. Technical Framework
3. Risk-Informed Baseline Inspection Program
4. Assessment Process
5. Enforcement Program Changes
6. Transition Plan
7. Summary of IRAP Public Comment
8. Commitments

KEY FIGURES AND TABLES

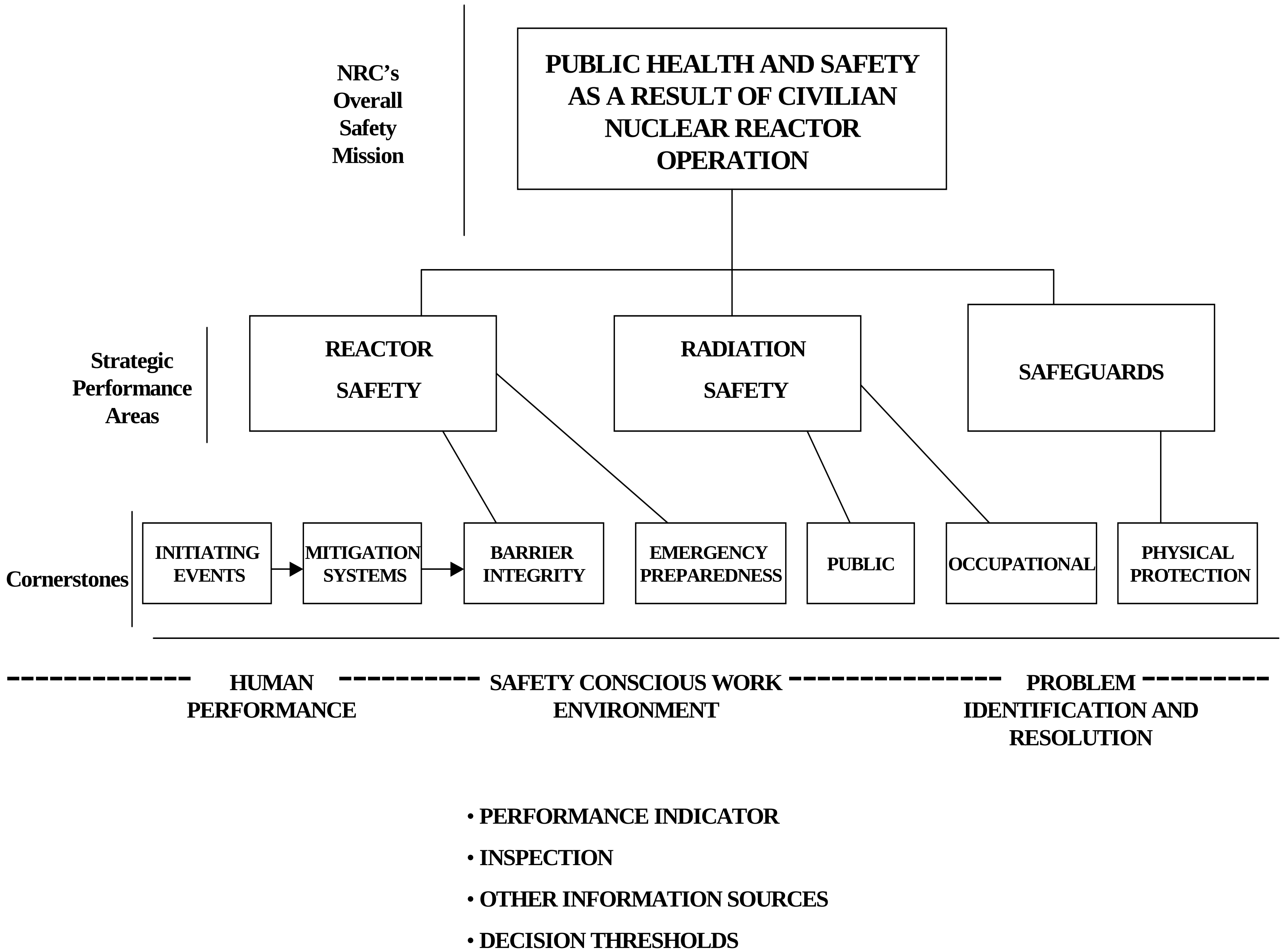


Figure 1 - Regulatory Oversight Framework

Table 1 - PERFORMANCE INDICATORS

Cornerstone	Indicator		Thresholds		
			Increased Regulatory Response Band	Required Regulatory Response Band	Unacceptable Performance Band
Initiating Events	Unplanned scrams per 7000 critical hours (automatic and manual scrams)		>3	>6	>25
	Risk-significant scrams per 3 years		>4	>10	>20
	Transients per 7000 critical hours		>8	N/A	N/A
Mitigating Systems	Safety System Performance Indicator Unavailability	HPCI and RCIC HPCS Emergency Power RHR AFW HPSI	>0.04 >0.015 >0.025 >0.015 >0.02 >0.015	>0.12 >0.04 >0.05 (>2EDG >0.1) >0.05 >0.06 >0.05	>0.5 >0.2 >0.1 (>2EDG >0.2) TBD >0.12 TBD
	Safety System Failures		>5 - prior 4 qtrs	N/A	N/A
Barriers **- Fuel Cladding**	Reactor coolant system (RCS) specific activity		>50% of TS limit	>100% of TS limit	N/A
- Reactor Coolant System	RCS leak rate		>50% of TS limit	>100% of TS limit	N/A
- Containment	Containment leakage		>100% L_A	N/A	N/A
Emergency Preparedness	Emergency Response Organization (ERO) drill/exercise performance		<75% - prior 6 months; <90% - prior 2 years	<55% - prior 6 months; <70% - prior 2 years	N/A
	ERO readiness (percentage of ERO shift crews that have participated in a drill or exercise in the past 24 months)		<80% - prior 2 years; <90% - prior 3 years	<60% - prior 2 years; <70% - prior 3 years	N/A

Table 1 - PERFORMANCE INDICATORS

Cornerstone	Indicator	Thresholds		
		Increased Regulatory Response Band	Required Regulatory Response Band	Unacceptable Performance Band
	Alert and Notification System performance (percentage of availability time)	<94% per year	<90% per year	N/A
Occupational Radiation Safety	Occupational exposure control effectiveness (the number of non-compliances with 10 CFR 20 requirements for (1) high (greater than 1000 mRem/hour) and (2) very high radiation areas, and uncontrolled personnel exposures exceeding 10% of the stochastic or 2% of the non-stochastic limits)	6 or more occurrences in 3 years (rolling average); 3 or more in 1 year	12 or more occurrences in 3 years (rolling average); 6 or more in 1 year	N/A
Public Radiation Safety	Offsite release performance (number of effluent events that are reportable per 10 CFR 20, 10 CFR 50 Appendix I, Offsite Dose Calculation Manual, or Technical Specifications)	7 or more events in 3 years (rolling average); 4 or more events in 1 year	14 or more events in 3 years (rolling average); 8 or more events in 1 year	N/A
Physical Protection	Protected Area security equipment performance (availability of systems to perform their intended functions)	<95% per year	<85% per year	N/A
	Vital Area security equipment performance (availability of systems to perform their intended functions)	<95% per year	<85% per year	N/A
	Personnel screening process performance (acceptable implementation of the access authorization program)	3-5 reportable events	6 or more reportable events	N/A
	Personnel reliability program performance (acceptable implementation of the fitness-for-duty & behavior observation programs)	3-5 reportable events	6 or more reportable events	N/A

CONCEPTUAL MODEL FOR EVALUATING LICENSEE PERFORMANCE INDICATIONS

- GREEN -
(ACCEPTABLE PERFORMANCE - Licensee Response Band)

-- Cornerstone objectives fully met
-- Nominal Risk/Nominal Deviation From Expected Performance

- WHITE -
(ACCEPTABLE PERFORMANCE -- Increased Regulatory Response Band)

-- Cornerstone objectives met with minimal reduction in safety margin
-- Outside bounds of nominal performance
-- Within Technical Specification Limits
-- Changes in performance consistent with ΔCDF<E-5 (ΔLERF<E-6).

- YELLOW -
(ACCEPTABLE PERFORMANCE - Required Regulatory Response Band)

-- Cornerstone objectives met with significant reduction in safety margin
-- Technical Specification limits reached or exceeded
-- Changes in performance consistent with ΔCDF<E-4 (ΔLERF<E-5)

- **RED** -
(UNACCEPTABLE PERFORMANCE - Plants not normally permitted to operate within this band)

-- Plant performance significantly outside design basis
-- Loss of confidence in ability of plant to provide assurance of public health and safety with continued operation
-- Unacceptable margin to safety

- U N S A F E P E R F O R M A N C E -

Inspectable Areas by Cornerstone

The baseline inspection program requires the inspectable areas below be reviewed at each nuclear power plant each year. The inspectable areas verify aspects of key attributes for each of the associated cornerstones. The inspectable areas are characterized as one of three basic types of inspection of cornerstones: complementary inspection, supplementary inspection, or verification inspection. Complementary inspections verify performance in areas that are not measured by a performance indicator, supplementary inspections augment the information provided by performance indicators that do not sufficiently measure performance in a cornerstone area; and verification inspections verify the accuracy and completeness of the data used as the basis for performance indicators used to fully measure performance of a cornerstone area.

- <u>Initiating Events Cornerstone</u>

  Adverse weather preparations (complementary)
  Equipment alignment (supplementary)
  Emergent work (complementary)
  Fire protection (complementary)
  Flood protection measures (complementary)
  Heat sink performance (complementary)
  Identification and resolution of problems and issues (complementary)
  Inservice inspection activities (complementary)
  Maintenance rule implementation (supplementary)
  Maintenance work prioritization and control (supplementary)
  Nonroutine plant evolutions (supplementary)
  Piping system erosion and corrosion (complementary)
  Refueling and outage activities (complementary)

- <u>Mitigating Systems Cornerstone</u>

  Adverse weather preparations (complementary)
  Changes to license conditions and safety analysis report (complementary)
  Emergent work (complementary)
  Equipment alignment (supplementary)
  Fire protection (complementary)
  Flood protection measures (complementary)
  Heat sink performance (complementary)
  Identification and resolution of problems and issues (complementary)
  Inservice testing of pumps and valves - ASME Section XI (complementary)
  Licensed operator requalification (complementary)
  Maintenance rule implementation (supplementary)
  Maintenance work prioritization and control (supplementary)
  Nonroutine plant evolutions (supplementary)
  Operability evaluations (complementary)
  Operator work-arounds (complementary)
  Permanent plant modifications (complementary)
  Post maintenance testing (supplementary)

Inspectable Areas by Cornerstone (continued)

Refueling and outage activities (complementary)
Safety system design and performance capability (complementary)
Surveillance testing (supplementary)
Temporary plant modifications (complementary)

- Barrier Integrity Cornerstone

  Changes to license conditions and safety analysis report (complementary)
  Equipment alignment (supplementary)
  Fuel barrier performance (verification)
  Identification and resolution of problems and issues (complementary)
  Inservice inspection activities (complementary)
  Large containment isolation valve leak rate and status verification (verification)
  Licensed operator requalification (complementary)
  Maintenance rule implementation (supplementary)
  Maintenance work prioritization control (supplementary)
  Nonroutine plant evolutions (supplementary)
  Permanent plant modifications (complementary)
  Refueling and outage activities (complementary)
  Surveillance testing (supplementary)
  Temporary plant modifications (complementary)

- Emergency Preparedness Cornerstone

  Alert and notification system testing (verification)
  Drill and exercise inspection (verification)
  Emergency action level changes (complementary)
  Emergency response organization augmentation testing (complementary)
  EP training program (verification)
  Identification and resolution of problems and issues (complementary)

- Occupational Exposure Cornerstone

  Access control to radiologically significant areas (supplementary)
  ALARA planning and controls (complementary)
  Identification and resolution of problems and issues (complementary)
  Radiation monitoring instrumentation (complementary)
  Radiation worker performance (complementary)

- Public Exposure Cornerstone

  Gaseous- and liquid-effluent treatment systems (supplementary)
  Identification and resolution of problems and issues (complementary)
  Radioactive-material processing and shipping (complementary)
  Radiological environmental monitoring program (complementary)

Inspectable Areas by Cornerstone (continued)

- Physical Security Cornerstone

  Access authorization (supplementary)
  Access control (complementary)
  Changes to license conditions and safety analysis report (complementary)
  Identification and resolution of problems and issues
  Physical protection system (verification)
  Response to contingency events (complementary)

REVIEW SYSTEM

Level of Review	**Frequency/ Timing**	**Participants (* indicates lead)**	**Desired Outcome**	**Communication**
Continuous	Continuous	SRI*, RI, regional inspectors, analysts	Performance awareness	None required
Quarterly	Once per quarter/ Two weeks after end of quarter	DRP: BC*, PE, SRI, RI	Input/verify PI/PIM data, detect early trends	Updated data set
Mid-Cycle	At mid-cycle/ Three weeks after end of second quarter	Divisions of Reactor Safety (DRS) or DRP DD*, DRP and DRS BCs	Detect trends, plan inspection for six months	Six month inspection look ahead letter
End-of-Cycle	At end-of-cycle/ Four weeks after end of assessment cycle	DRS or DRP DD*, RAs, NRR representative, BCs, principal inspectors, OE, OI, other HQ offices as appropriate	Assessment of plant performance, approve/ coordinate regional actions	Assessment letter and six month inspection look ahead letter
Agency Action Review	Annually/ Two weeks after end-of-cycle review	DIR NRR*, RAs, DRS/DRP DDs, AEOD, DISP, OE, OI, other HQ offices as appropriate	Approve/ coordinate agency actions	Commission briefing, followed by public meetings with individual licensees to discuss assessment results

Acronyms

SRI Senior Resident Inspector
RI Resident Inspector
BC Branch Chief
PE Project Engineer
DRP Division of Reactor Projects
DD Division Director
RA Regional Administrator
DIR Director
DISP Division of Inspection and Support Programs
OI Office of Investigations

		LICENSEE PERFORMANCE INCREASING SAFETY SIGNIFICANCE ---------->				
RESULTS		I. All Assessment Inputs (PIs and Cornerstone Inspection Areas) Green; Cornerstone Objectives Fully Met	II. One or Two Inputs White (in different cornerstones); Cornerstone Objectives Fully Met	III. One Degraded Cornerstone (2 Inputs White or 1 Input Yellow) or any 3 White Inputs; Cornerstone Objectives Met with Minimal Reduction in Safety Margin	IV. Repetitive Degraded Cornerstone, Multiple Degraded Cornerstones, or Multiple Yellow Inputs; Cornerstone Objectives Met with Significant Reduction in Safety Margin	V. Overall Red (Unacceptable) Performance; Plants Not Normally Permitted to Operate Within this Band, Unacceptable Margin to Safety
RESPONSE	Management Meeting	Routine Resident Inspector Interaction	SRI/BC Meet with Licensee	DD/RA Meet with Licensee Management	EDO Meet with Senior Licensee Management	Commission meeting with Senior Licensee Management
	Licensee Action	Licensee Corrective Action	Licensee Corrective Action with NRC Oversight	Licensee Self Assessment with NRC Oversight	Licensee Performance Improvement Plan with NRC Oversight	
	NRC Inspection	Risk-Informed Baseline Inspection Program	Inspection Follow-up	Inspection Focused on Cause of Degradation	Team Inspection Focused on Cause of Overall Degradation	
	Regulatory Actions	None	-Document Response to Degrading Area in Inspection Report	-Docket Response to Degrading Condition *(Consider N+1 Inspection for 2 Consecutive Cycles in This Range)*	-10 CFR 50.54(f) Letter - CAL/Order *(Consider N+1 Inspection for 2 Consecutive Cycles in This Range)*	Order to Modify, Suspend, or Revoke Licensed Activities
COMMUNICATION	Assessment Report	DD review/sign assessment report (w/ inspection plan)	DD review/sign assessment report (w/ inspection plan)	RA review/sign assessment report (w/ inspection plan)	RA review/sign assessment report (w/ inspection plan)	RA review/sign assessment report (w/ inspection plan)
	Public Assessment Meeting	SRI or Branch Chief Meet with Licensee	SRI or Branch Chief Meet with Licensee	RA Discuss Performance with Licensee	EDO Discuss Performance with Senior Licensee Management	Commission Meeting with Senior Licensee Management to Discuss Licensee Performance
		<---------- Regional Review I Agency Review ---------->				

Table 5 - Action Matrix

Technical Framework
For Licensee Performance Assessment

Team Leader	**Patrick Baranowsky**
Team Members	**Douglas Coe**
	John Flack
	Donald Hickman
	Jeffrey Jacobson
	William Johnson
	George Kuzo
	Scott Morris
	Robert Palla
	Gareth Parry
	Ann Ramey-Smith
	William Ruland
	Gregory Smith

Randolph Sullivan
John Wilcox

Attachment 2

Table of Contents

Page

Executive Summary

The NRC conducted a Performance Assessment Workshop on September 28 through October 1, 1998. The stated purpose of the workshop was to explore and develop a framework for oversight of operating commercial nuclear reactors. As an outcome of the workshop, alignment was reached on the general principles and conceptual framework that provides for a graded, threshold approach to performance assessment and related inspection. Subsequently, three task groups were formed to further develop the concept and make the NRC's reactor assessment and inspection processes more efficient, effective, and risk informed: (1) a technical framework task group; (2) an inspection task group; and (3) and assessment process task group. The task groups were comprised of representatives from the Office of Nuclear Reactor Regulation, the Office for the Analysis and Evaluation of Operational Data, the Office of Nuclear Regulatory Research, and all four NRC Regional Offices. This report provides the results of the technical framework task group's efforts to identify and develop:

- the cornerstones of safety and the key attributes of performance within each cornerstone;
- the performance indicators that can be used to assess performance in certain areas;
- performance indicator thresholds intended to establish clear demarcation points for identifying fully acceptable, declining, and unacceptable levels of performance;
- aspects of risk informed inspections that should supplement and verify the validity of the performance indicator data.

The task group also performed analysis of cross-cutting issues, benchmarked the proposed performance indicators against prior plant performance, and identified future development activities. During the development of the report, information was shared with the inspection and assessment process task groups for use in developing a new risk-informed baseline inspection program and overall NRC reactor assessment process.

As a starting point, the technical framework task group used the results of the Performance Assessment Public Workshop held from September 28 through October 1, 1998. During this workshop, alignment was reached with the industry on the regulatory oversight framework and the cornerstones of safety. A diagram of this framework showing the relationship between the NRC's overall safety mission, strategic performance areas, and cornerstones of safety is included as Figure 1 in the main body of the report.

These cornerstones of safety were chosen to: (1) limit the frequency of initiating events; (2) ensure the availability, reliability, and capability of mitigating systems; (3) ensure the integrity of the fuel cladding, reactor coolant system, and containment boundaries; (4) ensure the adequacy of the emergency preparedness functions; (5) protect the public from exposure to radioactive material releases;

(6) protect nuclear plant workers from exposure to radiation; and (7) provide assurance that the physical protection system can protect against the design basis threat of radiological sabotage.

Within each cornerstone area, the task group then used a top-down, risk-informed approach to:

- identify the objective and scope of the cornerstone;
- identify the desired results and important attributes of the cornerstone;
- identify what should be measured to ensure that the cornerstone objectives are met;
- determine which of the areas to be measured can be monitored adequately by performance indicators;
- determine whether inspection or other information sources are needed to supplement the performance indicators, and
- determine the thresholds of performance for each cornerstone, below which additional NRC actions would be taken.

Where possible, the task group sought to identify performance indicators as a means of measuring the performance of key attributes in each of the cornerstone areas. Where such a performance indicator could not be identified, the group proposed a “complementary” inspection activity. Where a performance indicator was identified but was not sufficiently comprehensive, the group proposed “supplementary” inspection activities. The task group also identified the need for “verification” type inspections to verify the accuracy and completeness of the reported performance indicator data. These recommended inspection activities were provided to the risk-informed baseline inspection task group for consideration in developing the baseline inspection program.

Performance indicators together with risk-informed baseline inspections, are intended to provide a broad sample of data to assess licensee performance in the risk significant areas of each cornerstone. They are not intended to provide complete coverage of every aspect of plant design and operation. It is recognized that licensees have the primary responsibility for ensuring the safety of the facility. Objective performance evaluation thresholds are intended to be used to help determine the level of regulatory engagement appropriate to licensee performance in each cornerstone area. Furthermore, based on past experience it is expected that a limited number of risk-significant events will continue to occur with little or no indication of declining performance. Follow up inspections will be conducted to ensure that the cause of the event is well understood and licensee corrective actions are adequate to prevent recurrence. The results of these follow-up inspections will be factored into the assessment process along with performance indicators and risk informed baseline inspections.

For the initiating events cornerstone, scrams per 7000 critical hours and transients were identified as performance indicators. Recommended inspection areas for this cornerstone included aspects of: fire protection; testing of steam generator tubes and reactor coolant system piping; and operating equipment line-ups.

Under mitigating systems, safety system failures and the safety system performance indicator (SSPI) were chosen as the performance indicators. Recommended inspection areas for this cornerstone included risk significant aspects of: protection of equipment from external events; equipment design adequacy and design modifications; test procedure adequacy; operator training/certification; and emergency operating procedures.

For the barrier integrity cornerstone, performance indicators were chosen for reactor coolant system activity, reactor coolant system leak rate, and total leakage from all containment penetrations. Recommended risk-informed inspections under this cornerstone included: configurations of control rod alignments during risk significant evolutions; configurations of key equipment in the reactor coolant system during shutdown; in-service inspection programs; equipment design adequacy and design modifications; and line-up of equipment penetrations.

The performance indicators selected for the emergency preparedness cornerstone were drill/exercise performance, emergency response organization readiness, and availability of the alert and notification system. Recommended inspection in the emergency preparedness area was largely centered around ensuring the adequacy of licensee assessments of exercises, drills, severe accident management guidelines, equipment, and facilities. In addition, inspection was recommended for changes to emergency action levels in accordance with 10 CFR 50.54(t) as appropriate.

In the area of Radiation Safety - Occupational Exposure, a summary performance indicator was crafted for occupational dose control. Inspection in this area was recommended for the identification and monitoring of high radiation areas; source term control; ALARA planning; and health physics technician performance.

In the area of Radiation Safety - Public Exposure, a performance indicator was chosen for effluent releases. Inspection was recommended for calibrations of and modifications to waste processing equipment; verifying operability of meteorological instrumentation; packaging and transportation of radioactive materials; and effluent sampling.

For the Physical Security cornerstone, performance indicators were selected for availability of security systems and failures of the personnel screening and fitness for duty process. Inspection in the Physical Security cornerstone was recommended for testing of barrier intrusion, detection, and alarm systems; search, identification, and control processes; response to security related incidents; and reporting of significant events.

A complete listing of the performance indicators selected for each cornerstone, along with performance thresholds is provided in Table 2 in the main body of this report. These thresholds were selected for consistency with the performance threshold conceptual model provided in figure 2 of the report. They correspond to levels of performance requiring no additional regulatory oversight (above the green to white threshold), performance that may result in increased oversight (below the green to white threshold), performance that will result in specific NRC actions (below the white to yellow threshold), and performance that is unacceptable (below the yellow to red threshold). It should be noted that although not expected, should a licensee's performance reach what has been determined to be an unacceptable level, margin would still exist before an undue risk to public health and safety would be presented. The extent of NRC actions would be graded based upon the relative deviation from the performance indicator threshold and the number of thresholds exceeded.

For some indicators, such as those for scrams and safety system performance indicators (SSPIs), selection of the performance indicator thresholds was made using the insights from probabilistic risk assessment (PRA) sensitivity analysis. Other performance indicator thresholds could not be assessed using PRA models. In such cases, the performance indicator thresholds were tied to regulatory requirements or were based on the professional judgement of the NRC staff and industry. For example, under the barrier integrity cornerstone, reactor coolant system activity is a good measure of the integrity of the fuel cladding, but the performance thresholds chosen were based on technical specifications. Under the physical security cornerstone, the availability of physical protection systems provides a useful measure of the status of intrusion detection equipment, but its thresholds were chosen based on professional judgement of the NRC staff and industry representatives.

Once the performance indicators and corresponding thresholds were selected, the task group performed a benchmarking analysis to compare the indicators against several plants that had been previously designated by the agency as having either poor, declining, average, or superior performance. The analysis indicated that the performance indicators could generally differentiate between poor and superior plants, but were not as effective at differentiating average levels of performance. The transients and safety system failure performance indicators appeared to be the most closely tied with prior NRC judgements about performance. In some instances, the cause of the plants rated poorly by the agency was due to design or other issues for which valid performance indicators have not been developed. It is expected that these plants would continue to be identified by the inspection program.

The task group also identified aspects of licensee performance such as human performance, the establishment of a safety conscious work environment, common cause failure, and the effectiveness of licensee problem identification and corrective action programs, that are not identified as specific cornerstones, but are important to meeting the safety mission. The task group concluded that these items generally manifest themselves as the root causes of performance problems. Adequate licensee performance in these crosscutting areas will be assessed either explicitly in each cornerstone area or will be inferred through cornerstone performance results from both PIs and inspection results.

1. Introduction

The NRC conducted a Performance Assessment Workshop on September 28 through October 1, 1998. The stated purpose of the workshop was to explore and develop a framework for oversight of operating commercial nuclear reactors. As an outcome of the workshop, alignment was reached on the general principles and conceptual framework that provides for a graded, threshold approach to performance assessment and related inspection. Subsequently, three task groups were formed to further develop the concept and make the NRC's reactor assessment and inspection processes more efficient, effective, and risk informed. The three task groups included: (1) a technical framework task group which was responsible for identifying what information is needed by the NRC in order to ensure adequate public health and safety, including whether the information could be obtained by performance indicators or whether inspections would be required; (2) an inspection task group which was responsible for defining the baseline inspection to be performed at all operating reactor facilities; and (3) and assessment process task group which was responsible for developing a process for assessing licensee performance using the performance indicator and inspection data. This report details the results of the technical framework task group. The purpose of the technical framework group was to further develop details of the conceptual framework derived from the workshop, namely:

- the cornerstones of safety and the key attributes of performance within each cornerstone;
- the performance indicators that can be used to assess performance in certain areas;
- performance indicator thresholds intended to establish clear demarcation points for identifying fully acceptable, declining, and unacceptable levels of performance;
- aspects of risk informed inspections that should supplement and verify the validity of the performance indicator data.

During the development of the report, information was shared with the inspection and assessment process task groups for use in crafting the baseline inspection program and the overall NRC reactor assessment process. The charter and roster of the technical framework group is provided as Appendix J to this report.

As a starting point, the technical framework task group used the results of the Performance Assessment Public Workshop in which the framework concept was developed. The cornerstone framework is a hierarchical structure that begins with a focus on the NRC's overall safety mission and identifies strategic areas in which performance must be maintained in order for the overall safety mission to be achieved. Each strategic performance area, in turn, has a set of cornerstones or safety function areas that support the strategic performance area. The cornerstones provide the fundamental building blocks for the regulatory oversight process and, if their objectives are met, provide reasonable assurance that the NRC's overall safety mission is also met. A diagram showing the NRC's overall safety mission, strategic performance areas, and cornerstones of safety is provided in Figure 1.

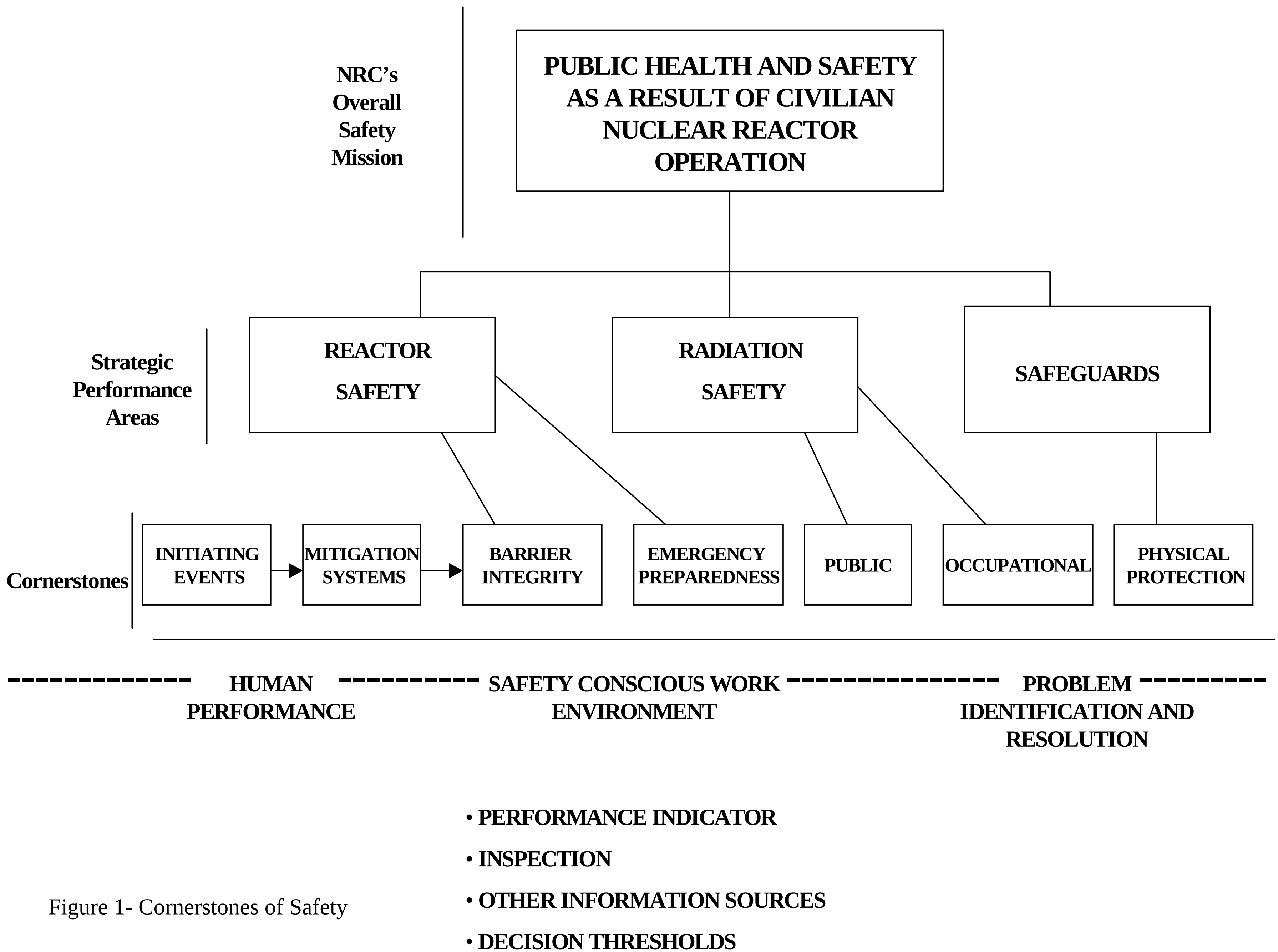


Figure 1- Cornerstones of Safety

For the reactor safety area, the cornerstones are:

Initiating Events - The objective of this cornerstone is to limit the frequency of those events that upset plant stability and challenge critical safety functions, during shutdown as well as power operations. If not properly mitigated and multiple barriers are breached, a reactor accident could result which would compromise the public health and safety. Licensees can reduce the likelihood of a reactor accident by maintaining a low frequency of these initiating events. Such events include reactor trips due to turbine trips, loss of feedwater, loss of off-site power, and other reactor transients.

Mitigating Systems - The objective of this cornerstone is to ensure the availability, reliability, and capability of systems that mitigate initiating events to prevent reactor accidents. Licensees reduce the likelihood of reactor accidents by enhancing the availability and reliability of mitigating systems. Mitigating systems include those systems associated with safety injection, residual heat removal, and their support systems, such as emergency AC power. This cornerstone includes mitigating systems that respond to both operating and shutdown events.

Barrier Integrity - The objective of this cornerstone is to ensure that physical barriers protect the public from radionuclide releases caused by accidents. Licensees can reduce the effects of reactor accidents or events if they do occur by maintaining the integrity of the barriers. The barriers are the fuel cladding, reactor coolant system boundary, and the containment.

Emergency Preparedness - The objective of this cornerstone is to ensure that actions taken by the emergency plan would provide adequate protection of the public health and safety during a radiological emergency. Licensees can ensure that the emergency plan would be implemented correctly by drills and training. This would give reasonable assurance that the licensee can effectively protect the public health and safety in the event of a radiological emergency. This cornerstone does not include the off-site actions, which are covered by FEMA.

For the reactor safety area to fail to meet the goal of adequate protection of public health and safety, an initiating event would have to occur, followed by failures in one or more mitigating systems, and ultimately failure of multiple barriers. At that stage, the emergency plan would be implemented as the last defense-in-depth of public protection.

For the radiation safety area, the cornerstones are:

Public Protection - The objective of this cornerstone is to ensure adequate protection of *public* health and safety from exposure to radioactive material released into the public domain as a result of *routine* civilian nuclear reactor operations. These releases include routine gaseous and liquid radioactive effluent discharges, the inadvertent release of solid contaminated materials, and the offsite transport of radioactive materials and wastes. Licensees can maintain public protection by meeting the applicable regulatory limits and ALARA guidelines.

Occupational Worker Protection - The objective of this cornerstone is to ensure adequate protection of *worker* health and safety from exposure to radiation from radioactive material during *routine* civilian nuclear reactor operation. This exposure could come from poorly controlled or uncontrolled radiation areas or radioactive material that unnecessarily exposes workers. Licensees can maintain occupational worker protection by meeting applicable regulatory limits and ALARA guidelines.

For safeguards, the cornerstone is:

Physical Protection - The objective of this cornerstone is to provide assurance that the physical protection system can protect against the design basis threat of radiological sabotage. The threat could come from either external or internal threats. Licensees can maintain adequate protection against threats of sabotage based on an effective security program that relies on a defense in depth approach.

Performance indicators together with risk-informed baseline inspections, are intended to provide a broad sample of data to assess licensee performance in the risk significant areas of each cornerstone. They are not intended to provide complete coverage of every aspect of plant design and operation. It is recognized that licensees have the primary responsibility for ensuring the safety of the facility. Objective performance evaluation thresholds are intended to be used to help determine the level of regulatory engagement appropriate to licensee performance in each cornerstone area. Furthermore, based on past experience it is expected that a limited number of risk significant events will continue to occur with little or no prior performance indication. Reactive inspections will be conducted to ensure that the cause of the event is well understood and licensee corrective actions are adequate to prevent recurrence. The results of these follow-up inspections will be factored into the assessment process along with performance indicators and risk informed baseline inspections.

2. Framework Development Process

The Performance Assessment Workshop began the process of identifying the key attributes of performance within each cornerstone area, as well as potential performance indicators. The framework task group continued the process using a top-down approach to developing each cornerstone area. Included within the task group's efforts were:

- defining the objective and scope of the cornerstone;

- identifying the desired results for each cornerstone and the important licensee performance attributes necessary to achieve them;

- identifying what attributes of performance the NRC needs to assess to ensure that the cornerstone objectives are met;

- determining which of the attributes to be monitored can be measured adequately by performance indicators;

- determining whether inspection or other information sources are needed to supplement the performance indicators, and

- determining the thresholds of performance for each cornerstone, above which additional NRC actions would be taken.

A detailed analyses of each cornerstone is provided in Appendices A through G. Throughout the task group's efforts, a risk informed approach was used, meaning that probabilistic risk insights were balanced with operational experience and existing regulations. In general, generic versus site specific probabilistic risk assessment results were used in selecting the performance indicators, performance thresholds, and inspection areas. This generic use of risk information is consistent with previous NRC risk-informed applications such as guidance to the Maintenance Rule and Regulatory Guide 1.174, “Scope, Level of Detail and Quality of PRA.”

Where possible, the task group sought to identify performance indicators as a means of measuring the performance of key attributes in each of the cornerstone areas. In selecting performance indicators, the task group tried to select indicators that: (1) were capable of being objectively measured; (2) allowed for the establishment of a risk-informed threshold to guide NRC and licensee actions; (3) provided a reasonable sample of performance in the area being measured; (4) represented a valid and verifiable indication of performance in the area being measure; (5) would encourage appropriate licensee and NRC actions; and (6) would provide sufficient time for the NRC and licensees to correct performance deficiencies before the deficiencies posed an undue risk to public health and safety. Where such a performance indicator could not be identified, the group proposed a “complementary” inspection activity. Where a performance indicator was identified but was not sufficiently comprehensive to cover all performance areas to be measured, the group proposed “supplementary” inspection activities. The task group also identified areas where “verification” type inspections should be performed to verify the accuracy and completeness of the reported performance indicator data.

In some instances, performance indicator thresholds could be directly tied to probabilistic risk assessment data, such as those for scrams and safety system performance indicators (SSPIs) (See Appendix H). A sample of plants with PRA models available was selected to cover a spectrum of “typical” designs. Normal performance ranges were identified and core damage frequency sensitivity analyses were performed to evaluate the effects of departures from normal performance. This information was used to set performance indicator threshold values that corresponded to the nominal (acceptable), declining (acceptable), and unacceptable performance bands described in section 3.

Other performance indicator thresholds could not be specifically tied to probabilistic risk data. In such cases, the performance indicator thresholds were tied to regulatory requirements or were based on the professional judgement of the NRC staff and industry. For example, under the barrier integrity cornerstone, reactor coolant system activity is a good measure of the integrity of the fuel cladding, but the performance thresholds chosen were based on technical specifications. Under the physical security cornerstone, the availability of physical protection systems provides a useful measure of the status of intrusion detection equipment, but its thresholds were chosen based on professional judgement of the NRC staff and industry representatives. It is expected that if a licensee was to exceed a performance indicator threshold, additional NRC actions, including inspection, would be taken in order to identify the cause and prevent any undue risk to public health and safety. The extent of NRC actions would be graded based upon the relative deviation from the performance indicator threshold and the number of thresholds exceeded.

The task group also identified aspects of licensee performance such as human performance, the establishment of a safety conscious work environment, common cause failure, and the effectiveness of licensee problem identification and corrective action programs, that are not identified as specific cornerstones, but are important to meeting the safety mission. The task group concluded that these items generally manifest themselves as the root causes of performance problems.   Adequate licensee performance in these crosscutting areas will be assessed either explicitly in each cornerstone area or will be inferred through cornerstone performance results from both PIs and inspections.   A more detailed discussion of cross-cutting issues and how they are specifically addressed is discussed in section 6 of this report.

Lastly, the selected PIs were put through a benchmarking exercise that involved evaluation of an industry sponsored assessment and independent NRC staff analyses. This benchmarking was performed for a selection of plants with a history of poor, declining, average, and superior performance as determined by the NRC's senior management meetings. (See Appendix I)

3. Performance Thresholds Conceptual Framework

The concept for setting performance thresholds includes consideration of risk and regulatory response to different levels of licensee performance. The approach is intended to be consistent with other NRC risk-informed regulatory applications and policies as well as consistent with regulatory requirements and limits. The primary attributes of the concept are: (1) the scheme should include multiple levels with clearly defined thresholds to allow unambiguous observation and assessment of declining (or improving) performance; (2) the thresholds should be risk informed to the extent practical, but should accommodate defense in depth and indications based on existing regulatory requirements and safety analyses; (3) the risk implications and regulatory actions associated with each performance band and associated threshold should be consistent with other NRC risk applications, and based on existing criteria where possible (e.g. Regulatory Guide 1.174) ; (4) the scheme should provide for consistency of risk informed indications of performance with performance indications based on existing regulatory requirements and safety analyses to the extent practical; (5) the scheme should be capable of accounting for performance indicated by risk-informed inspection findings; (6) thresholds should provide sufficient differential to allow meaningful differentiation in performance and limit false positives (e.g. allow an order of magnitude in the risk differential between thresholds); (7) sufficient margin should exist between nominal performance bands to allow for licensee initiatives to correct performance problems before reaching escalated regulatory involvement thresholds, and sufficient margin should exist between thresholds that signify initial declining performance and unacceptable performance to allow for both NRC and licensee diagnostic and corrective actions to be effectuated; (8) each individual PI should have its own performance thresholds; (9) where appropriate plant-specific design differences should be accommodated; and (10) there will be a performance threshold for unacceptable performance sufficiently above the point of unsafe plant operation that allows NRC sufficient opportunity to take appropriate action to preclude operation in this condition.

The conceptual model that was developed to incorporate the attributes listed above is shown in figure 2. It includes four performance bands and their general performance characteristics as discussed below:

- The licensee response band is characterized by acceptable performance in which cornerstone objectives are being met with performance attributes and risk indications of individual performance assessment indications (PIs and inspection findings)

Figure 2 CONCEPTUAL MODEL FOR EVALUATING LICENSEE PERFORMANCE INDICATIONS

- GREEN -
(ACCEPTABLE PERFORMANCE - Licensee Response Band)

-- Cornerstone objectives fully met
-- Nominal Risk/Nominal Deviation From Expected Performance

- WHITE -
(ACCEPTABLE PERFORMANCE -- Increased Regulatory Response Band)

-- Cornerstone objectives met with minimal reduction in safety margin
-- Outside bounds of nominal performance
-- Within Technical Specification Limits
-- Changes in performance consistent with ΔCDF<E-5 (ΔLERF<E-6).

- YELLOW -
(ACCEPTABLE PERFORMANCE - Required Regulatory Response Band)

-- Cornerstone objectives met with significant reduction in safety margin
-- Technical Specification limits reached or exceeded
-- Changes in performance consistent with ΔCDF<E-4 (ΔLERF<E-5)

- **RED** -
(UNACCEPTABLE PERFORMANCE - Plants not normally permitted to operate within this band)

-- Plant performance significantly outside design basis
-- Loss of confidence in ability of plant to provide assurance of public health and safety with continued operation
– Unacceptable margin to safety

- U N S A F E P E R F O R M A N C E -

in the normal range. This performance band is also designated as the green band. Performance problems would not be of sufficient significance that escalated NRC engagement would occur. Licensees would have maximum flexibility to "manage" corrective action initiatives. The threshold for this band would involve performance that would be outside the normal range of industry historical performance and risk.

- The increased regulatory response band would be entered when licensee performance is outside the normal performance range, but would still represent an acceptable level of performance. This performance band is also designated as the white band. Performance is still considered to be within the objectives of the cornerstone and is within TS limits, but there is indication of declining performance and reduced safety limits. Degradation in performance in this band is typified by changes in risk of up to $\Delta 10^{-5}$ CDF or $\Delta 10^{-6}$ LERF associated with either PIs or inspection findings. The CDF and LERF threshold characteristics were selected to be consistent with Regulatory Guide 1.174 applications. Currently only CDF has been factored into performance indicator thresholds (See Appendix H).

- The required regulatory response band involves more significant decline in performance but licensee performance is, in general, still considered acceptable, if marginal. Performance in this band is also designated as the yellow band. When TS limits are reached or exceeded, licensees would be required to take immediate and effective corrective actions to maintain performance in the band. Degradation in performance in this band is typified by changes in risk of up to $\Delta 10^{-4}$ CDF or $\Delta 10^{-5}$ LERF associated with either PIs or inspection findings. These threshold characteristics and implied regulatory response are also selected to be consistent with risk-informed regulatory applications and mandatory actions for regulatory compliance.

- The unacceptable performance band is entered when performance falls bellow the yellow band threshold. It is also designated the red band and is typified by changes in performance that are indicative of changes in risk greater than $\Delta 10^{-4}$ CDF or $\Delta 10^{-5}$ LERF associated with either PIs or inspection findings. Plant performance is considered to be significantly outside the design basis, with unacceptable margin(s) to safety, with an accompanied loss of confidence that public health and safety would be assured with continued operation. Further decline in performance would result in operation in a state inconsistent with the safety goals.

It is recognized that some aggregation of PI and/or inspection findings will be necessary to apply this threshold model to evaluate cornerstone performance implications. The proposed aggregation concept is provided in the report from the Assessment Process Task Group.

4. Performance Indicators

Twenty performance indicators were developed in support of the cornerstone approach to licensee performance assessment, with at least one PI established for each of the seven cornerstones. The *Safety System Performance Indicator (SSPI)* is actually four individual indicators to measure the availability of four different safety systems. Another PI, *Occupational Exposure Effectiveness*, is a composite indicator which sums occurrences in three areas to assess performance in the Occupational Radiation Safety cornerstone. Table 1 provides a listing of theses PIs and includes a brief definition of the specific data that will be collected for each performance indicator along with the performance thresholds.

In a September 10, 1998, white paper, the NEI proposed eleven PIs to assess licensee safety performance. In general, all of these PIs are encompassed by the set of indicators established in the cornerstone framework. The current proposed set of PIs includes four PIs to evaluate licensee security and safeguards practices for the physical protection cornerstone, an area not initially considered by NEI. Differences between the initial NEI PIs and the NRC PIs largely involve the scope of the areas monitored, as well as the thresholds for which regulatory response would commence. Licensees will record most PI data on a monthly basis, and report the indicators on a quarterly basis. For PIs that are measured at intervals greater than quarterly (such as containment leakage), data will be reported at the end of each quarter in which they are measured. Reporting of PIs to the NRC will be established by a voluntary process, which is needed to enable the cornerstone performance assessment process to be implemented in the relatively near term.

Several additional PIs have been proposed, however further work is needed to determine whether these proposed PIs are viable and can provide meaningful licensee performance insights. Additional discussion of these proposed indicators is detailed in section 4 as well as the individual cornerstone appendices.

For two PIs (transients and safety system failures), no thresholds have been identified for the Required Regulatory Response Band or the Unacceptable Performance Band because the indicators could not be directly tied to risk data. These two indicators have provided good correlation with plant performance in the past and they are considered to be leading indicators of the more risk-significant indicators (scrams, risk-significant scrams, and SSPI). The barrier integrity cornerstone PIs (RCS activity, RCS leak rate, and containment leakage) do not have thresholds identified for the Unacceptable Performance Band because their lower thresholds are based on regulatory requirements (technical specifications). Individual plant technical specifications would require plant shutdown within a short time after the regulatory limits were exceeded. The emergency preparedness, radiation safety, and safeguards cornerstones do not have thresholds identified for the Unacceptable Performance Band. There is no risk basis for a determination that a certain degraded level of performance reflected by these indicators can be correlated into mandatory plant shutdown. It is expected that declining performance in the areas monitored by these indicators would be arrested by increased licensee corrective actions and by increased NRC attention up to and including the issuance of orders.

5. Risk Informed Inspection Areas

As stated previously, the performance indicators do not cover the complete spectrum of risk significant attributes necessary to assess performance in each cornerstone area. In some cases, such as with the SSPIs, the indicators provide data for only a subset of risk significant systems and components. In other areas, such as with design adequacy, the performance indicators provide little or no data. Consequently, risk informed inspection is required to ensure that performance is adequate in certain areas. This inspection can be divided into three parts: (1) inspection required to supplement those areas where performance indicators do exist but are not sufficiently comprehensive; (2) complementary inspections required where indicators do

Table 1 - PERFORMANCE INDICATORS

Cornerstone	Indicator		Thresholds: Increased Regulatory Response Band	Thresholds: Required Regulatory Response Band	Thresholds: Unacceptable Performance Band
Initiating Events	Unplanned scrams per 7000 critical hours (automatic and manual scrams)		>3	>6	>25
	Risk-significant scrams per 3 years		>4	>10	>20
	Transients per 7000 critical hours		>8	N/A	N/A
Mitigating Systems	Safety System Performance Indicator Unavailability	HPCI and RCIC HPCS Emergency Power RHR AFW HPSI	>0.04 >0.015 >0.025 >0.015 >0.02 >0.015	>0.12 >0.04 >0.05 (>2EDG >0.1) >0.05 >0.06 >0.05	>0.5 >0.2 >0.1 (>2EDG >0.2) TBD >0.12 TBD
	Safety System Failures		>5 - prior 4 qtrs	N/A	N/A
Barriers **- Fuel Cladding**	Reactor coolant system (RCS) specific activity		>50% of TS limit	>100% of TS limit	N/A
- Reactor Coolant System	RCS leak rate		>50% of TS limit	>100% of TS limit	N/A
- Containment	Containment leakage		>100% L_A	N/A	N/A
Emergency Preparedness	Emergency Response Organization (ERO) drill/exercise performance		<75% - prior 6 months; <90% - prior 2 years	<55% - prior 6 months; <70% - prior 2 years	N/A

Table 1 - PERFORMANCE INDICATORS

Cornerstone	Indicator	Thresholds		
		Increased Regulatory Response Band	Required Regulatory Response Band	Unacceptable Performance Band
	ERO readiness (percentage of ERO shift crews that have participated in a drill or exercise in the past 24 months)	<80% - prior 2 years; <90% - prior 3 years	<60% - prior 2 years; <70% - prior 3 years	N/A
	Alert and Notification System performance (percentage of availability time)	<94% per year	<90% per year	N/A
Occupational Radiation Safety	Occupational exposure control effectiveness (the number of non-compliances with 10 CFR 20 requirements for (1) high (greater than 1000 mRem/hour) and (2) very high radiation areas, and uncontrolled personnel exposures exceeding 10% of the stochastic or 2% of the non-stochastic limits)	6 or more occurrences in 3 years (rolling average); 3 or more in 1 year	12 or more occurrences in 3 years (rolling average); 6 or more in 1 year	N/A
Public Radiation Safety	Offsite release performance (number of effluent events that are reportable per 10 CFR 20, 10 CFR 50 Appendix I, Offsite Dose Calculation Manual, or Technical Specifications)	7 or more events in 3 years (rolling average); 4 or more events in 1 year	14 or more events in 3 years (rolling average); 8 or more events in 1 year	N/A
Physical Protection	Protected Area security equipment performance (availability of systems to perform their intended functions)	<95% per year	<85% per year	N/A
	Vital Area security equipment performance (availability of systems to perform their intended functions)	<95% per year	<85% per year	N/A
	Personnel screening process performance (acceptable implementation of the access authorization program)	3-5 reportable events	6 or more reportable events	N/A

Table 1 - PERFORMANCE INDICATORS				
Cornerstone	**Indicator**	**Thresholds**		
		Increased Regulatory Response Band	Required Regulatory Response Band	Unacceptable Performance Band
	Personnel reliability program performance (acceptable implementation of the fitness-for-duty & behavior observation programs)	3-5 reportable events	6 or more reportable events	N/A

not exist, and (3) verification type inspection activities designed to ensure the completeness and accuracy of the reported performance indicator data.

Included within Appendices A-G are a complete listing of recommended inspection elements tied to each cornerstone, key attribute, and inspection area. Based on expert panel review, these inspection elements were determined to have a direct relationship to meeting the desired performance goals in each cornerstone area. These inspection recommendations were then compared against the preliminary results of the Risk-Informed Baseline Inspection Task Group. The resulting list of inspectable areas contained in the report from the Risk-Informed Baseline Task Group provides an integrated list of recommended inspection areas. Not included within the baseline inspection program are reactive inspections which may be performed in response to specific high risk events or in response to exceeding the thresholds associated with specific performance indicators.

6. CROSS-CUTTING ISSUES

Certain aspects of licensee performance were seen as “cross-cutting” and potentially impacting more than one cornerstone. Issues identified during the Performance Assessment Workshop included: (1) human performance, (2) establishment of a safety conscious work environment, and (3) the effectiveness of problem identification and corrective action programs. Three other closely related issues were identified by the framework task group and are included in the discussion below: (1) maintenance rule implementation; (2) common cause failure; and (3) generic issues and risk significant events. During the group’s efforts to assess the information needed to ensure adequate performance in each cornerstone area, the cross-cutting issues were considered and where possible, linked to either performance indicators or inspection areas. They are discussed below to characterize their significance and means by which they were addressed during the cornerstone development process.

Human Performance

By the nature of the design of nuclear power plants and the role of plant personnel in maintenance, testing, and operation, human performance plays an important role in normal, off-normal, and emergency operations. Following the accident at Three Mile Island, Unit 2 (TMI-2), the NRC implemented a number of programs that significantly improved the reliability of personnel performance and the safety of nuclear power plants by reducing the likelihood of core damage and containment failure. Detailed control room design reviews resulted in substantial improvements to the human engineering design of control rooms, as well as to control stations and panels outside the main control room. Emergency operating procedures were modified to include symptom-oriented mitigation strategies and were refined to be more useable, reducing errors in their implementation. Training programs for licensed operators, and later for other important plant personnel, were modified such that job-task analyses were performed which formed the basis for the development of learning objectives, training materials and approaches, objective-specific testing, and appropriate program improvements based on feedback from personnel performance in the field. Other policies and programs implemented by the NRC improved staffing, overtime controls, and fitness-for-duty of plant personnel. Still others improved security and safeguards operations, emergency planning and response, and health physics controls (both occupational and public). Broad-reaching verification and validation efforts were conducted to ensure the proper implementation of the programs. Together, these programs have significantly improved human performance.

Risk-informed, performance-based regulation will, at least in part, involve a shift in the NRC role from *improving* human reliability to one of *monitoring* human reliability. Past efforts were appropriately pro-active (rather than performance based) because the accident at TMI-2 had clearly illustrated the serious deficiencies in programs to support effective and safe human performance. The success of the human performance improvement programs allows the NRC to now take a more performance-based approach to regulatory oversight of human performance. Thus, if plant performance is acceptable (as monitored through risk-informed inspections and performance indicators), then the performance of plant personnel is assumed to be acceptable as well. That is, if risk-informed inspection (for example, maintenance rule verification inspections, configuration control inspections, and other inspections as described for each cornerstone) and plant performance indicators for each cornerstone (such as scrams and transients for the initiating events cornerstone and the SSPI for the mitigating systems cornerstone) together indicate that plant performance is meeting the cornerstone objectives, then those findings also provide an indication of the acceptability of the associated human activities. This relationship between plant and human performance is assumed to be especially strong with regard to the broad range of normal operations, including maintenance and testing activities during power and shutdown operations. Supplemental verification inspections of problem identification and resolution programs will be conducted to ensure that human performance (and those factors such as training, procedures, and the like that influence human performance) is specifically and appropriately investigated through licensees' root cause analyses and corrective action programs, including the investigation of potential common cause failures caused by human actions.

Post-initiator operator actions are far less frequent than pre-initiator human activities that influence the latent capability of plant equipment. While initial and requalification examinations provide a predictive measure of operator performance during off-normal and emergency operations, follow-up inspections of risk-significant events will provide a more direct indication of the adequacy of post-initiator human performance. In addition, performance measures from emergency response exercises, and those associated with security and occupational exposure, will provide another means for the NRC to ensure that human reliability is being maintained appropriately.

Safety Conscious Work Environment

A safety conscious work environment (SCWE), also referred to as a "safety culture," can be characterized by a willingness on the part of a licensee staff to raise and document safety issues to resolve risk-significant equipment and process deficiencies promptly, adhere to written procedures, conduct effective training, make conservative decisions, and conduct probing self-assessments. In general, management commitment to safety will promote a safety conscious work environment. Possible indications of an "unhealthy" safety culture include a high number of allegations, a weak employee concerns program, and a high corrective maintenance backlog.

The establishment of a safety conscious work environment is seen as a cross-cutting issue since a poor safety culture among licensee staff can affect performance in any of the cornerstone areas. For example, a failure to reinforce high standards of procedure compliance or provide effective training can result in human-induced errors which cause transient events or render safety systems inoperable (initiating events and/or the mitigating systems cornerstone). A corrective action program with a high threshold for reporting conditions adverse to quality can result in a large number of deficiencies going unresolved, which could complicate plant response to a subsequent event (mitigating systems or barriers cornerstone).

The importance of a safety conscious work environment is similar to, if not integral with, the role of licensee problem identification and corrective action processes. As with the problem identification and corrective action cross-cutting issue, an assumption was made regarding the role of a safety conscious work environment in NRC assessments of licensee performance. Specifically, if a licensee had a poor safety conscious work environment, problems and events would continue to occur at that facility to the point where either they would result in exceeding thresholds for various performance indicators, or they would be surfaced during NRC baseline inspection activities, or both. Additionally, because inspection of licensee problem identification and corrective action programs will be included in the baseline inspection program, some indirect assurance will be gained as to the health of a licensee's safety culture. Lastly, the NRC's verification of the maintenance rule implementation, also to be included in the baseline inspection program, will provide assurance that risk-significant safety equipment deficiencies are being effectively resolved. In short, no separate and distinct assessment of licensee safety culture is needed because it is subsumed by either the PI's or baseline inspection activities.

Problem Identification and Corrective Action Programs

Defining and implementing an effective problem identification and corrective action program is a key element underlying licensee performance in each cornerstone area. A fundamental goal of the NRC's reactor inspection and assessment process is to establish confidence that each licensee is detecting and correcting problems in a manner that limits the risk to members of the public. The NRC expects licensees to be technically and organizationally self-sufficient in this regard. Ineffective problem identification and corrective action programs, including poor conduct of root cause analysis of self-identified or self-revealing issues, has been a common theme among problem plants in the past. The scope of problem identification and corrective action programs includes processes for self-assessment, root cause analysis, safety committees, operating experience feedback, and corrective action.

With regard to licensee problem identification and corrective action effectiveness, there are several areas that are not specifically evaluated by either the individual cornerstone performance indicators or the complimentary risk-informed inspections. As such, additional focused inspection is needed to evaluate licensee performance as it relates to this cross-cutting issue. Specifically, baseline inspection of licensee corrective action programs is necessary for the NRC to:

(1) conduct reviews of precursors to events which occur relatively infrequently but have significant consequences;

(2) independently identify potentially "generic" concerns that a licensee may have missed, including specific problems involving safety equipment, procedure development, design control, etc.;

(3) assess the collective impact of all of the items in the corrective action backlog which may not have individual risk significance. The cornerstone framework does not otherwise include a means to accomplish this assessment. A good understanding of plant-specific risk vulnerabilities would be needed while conducting this review;

(4) have assurance that licensees adequately address potential “common cause” equipment failure concerns, identified either by internal events and issues or by receipt of operating experience feedback from other licensees, vendors, etc.;

(5) verify that licensee’s are appropriately identifying and capturing issues that could affect the unavailability of equipment tracked by the SSPIs and the maintenance rule.

In all cases, deficiencies identified in the problem identification and corrective action program should be risk informed and should be tied to the cornerstone areas.

Maintenance Rule Implementation

Assessment of the licensee’s implementation of the maintenance rule is recommended as a baseline inspection item in order to ensure the availability, reliability, and capability of those safety systems and components (SSCs) not being monitored by the safety system performance indicators (SSPIs). The maintenance rule includes SSCs that influence performance in the initiating events, mitigation systems, and barrier integrity cornerstone areas. SSCs under the scope of the maintenance rule include both safety related SSCs, and non-safety related SSCs. These SSCs are relied upon to mitigate accidents or transients and are used in plant emergency operating procedures (EOPs). Failure of the SSCs could also cause a reactor scram or actuate a safety-related system. The maintenance rule requires monitoring of the reliability and unavailability of risk significant SSCs, as well as the performance of on-line and shutdown safety assessments of equipment line-ups. These shutdown and on-line safety assessments are important in order to ensure that the defense-in-depth and safety margins features of the plant design are not unduly compromised.

This maintenance rule verification inspection activity should consist of two parts: (1) verification of the accuracy and completeness of the unreliability and unavailability data that provides input to individual performance indicators (PI); and (2) monitoring of the plant specific high risk SSCs that have demonstrated high unreliability and unavailability to ensure that appropriate corrective actions have been taken.

Common Cause Failure

A common cause failure (CCF) is an event or condition that results in the failure of redundant equipment to perform its safety function at approximately the same time as a result of a shared cause. Risk assessments have consistently shown that CCF is a significant contributor to risk. This is due in large part to the multiple redundancies in commercial nuclear power plant design that makes coincident independent failure of redundant systems and components relatively insignificant. Since common cause failures can impact the performance of systems in multiple cornerstones (or multiple systems within cornerstones) it is important for the performance indicators and risk-informed inspection activities to appropriately account for their impact.

Common cause failures in risk important systems tend to be relatively rare events that are not easily amenable to plant specific trending and monitoring. The rate of CCF events resulting in complete failure of redundant equipment has been steadily decreasing in recent years. The current rate is about one event at a plant every 5-6 years. The performance indicators and risk-informed inspections are designed to monitor performance of risk important systems, structures and components. Failures causing initiating events or the inability of a risk-significant mitigating system to perform its safety function would be captured under the existing process. Common cause failures, being a subset of all failures would be captured by this process as well. However, since a common cause failure has a greater potential risk impact, additional activity beyond monitoring the impact of failures on performance indicators and other baseline inspections is warranted.

The NRC has developed a CCF database which catalogs both the complete and partial common cause failure events that have occurred between 1980 and 1996 as reported in the LERs and INPO's NPRDS database (NPRDS is being replaced by EPIX which will provide input for future CCF event analyses). AEOD is currently performing a study of the nature of the causes, coupling factors, and barriers to common cause failures from the industry wide experience contained in the CCF database. Insights from this study and plant specific screening of the CCF data can be used to identify plant specific programmatic areas for inspection that will enable the NRC to assess whether the most important causes, coupling factors, and barriers to CCF are being addressed by licensees. With the existing coverage of failure events (including the CCF events) by the performance indicators and other risk-informed inspection activities, the addition of CCF insights to appropriate inspection activities can provide added assurance that licensee are adequately addressing factors important to limiting the occurrence of CCF events.

Generic Issues and Risk Significant Events

Generic issues are events or conditions affecting the safety performance of plant systems, structures, or components which have the potential to affect multiple plants. They may be identified as a result of an occurrence at a plant(s) or a review of one or more plant's design features. The NRC has a long standing program to identify generic issues, rank them according to their risk significance, and resolve them in accordance with their attendant costs and benefits. Generally, generic events deal with issues outside the existing licencing bases of the plants.

Another long standing and complementary program to the Generic Issue process is the evaluation of the risk significance of operational events in the Accident Sequence Precursor Program. This program calculates the conditional probability of core damage based on actual events or conditions reported at nuclear power plants. It provides the ability to relatively rank the severity of operational events and can also provide generic insights into plant and industry performance. In addition, the analysis of ASP events provides insights into the characteristics of the most risk significant event occurrences as well as trending industry experience relating to these events. SECY 97-296, SECY 98-298, and NUREG/CR-4674, Volume 26 provide more detail on these findings including:

- The rate of occurrence of precursors has been decreasing significantly
- The most risk significant precursors (CCDP>1E-3) tend to occur about once every two years
- The risk implication of the ASP events is generally consistent with estimates based on PRA/IPE analyses

- Of the approximately 1000 events and conditions reportable to the NRC each year, about 1% have sufficient risk implications (CCDP>1E-6) to be precursors

- The characteristics of about 15% of the precursors are different from those typically modeled in PRA/IPEs

Both the ASP insights provided above and the benchmarking of PIs against ASP events (see Appendix I) indicate that there will be limited leading indication for many of these events. However, as also noted in the ASP insights, the industry-wide occurrence rate of these events and conditions is generally consistent with estimates based on PRA and IPE results. While current PRAs, IPEs, and ASP analyses do not cover all possible contributors to risk, for the areas they do cover (primarily core damage events associated with “internal” initiators and some containment events), they indicate consistency with the quantitative health objectives of the Commission’s Safety Goal Policy. Moreover, the current NRC Strategic Plan has a performance goal to “Maintain low frequency of events which could lead to severe accidents”, which it further defines as an occurrence rate of not more than one per year of “events that could result in a 1/1000 (10^{-3}) or greater probability of occurrence of a severe accident”

As was also the case with CCF events, the risk importance of these events likely will be relatively higher than other events captured by the performance indicator or baseline inspections. The ASP analyses will continue to provide indication of the overall industry performance and the frequency at which these events are expected to occur as well as provide perspective on the most risk significant events at individual plants. This program and related activities that provide risk based analysis of reactor operating experience will continue to provide the industry-wide context and assessment of events and issues that affect individual plants and have broader risk significant implications.

7. Future Development Activities

Work is needed to complete several activities to further enhance the reactor performance assessment framework. A brief summary of these activities follows.

PI Definitions

- Detailed instructions for how to calculate each PI, must be established and distributed.

- A review of more optimal PI display options should be conducted, particularly for the SSPI and Barrier indicators.

PI Thresholds

- The current assessment of PI thresholds is based on a relatively small number of sensitivity studies, using PRA models of differing levels of detail. They show significant differences in results. The selected threshold values are somewhat conservative for most but not all plants. More effort is needed to understand these results, and to determine whether thresholds can or should be established for plant classes on a plant specific basis. In particular, the following activities should be performed:

- If made available, analyze the results of sensitivity studies performed/provided by NEI for a broader set of plants, conducted in accordance with the approach used by NRC staff.

- Separate the analysis of data on turbine driven pump and motor driven pump trains (contingent on receiving data from industry)

- Conduct analyses to support the establishment of thresholds for new/proposed PIs, or refinement of existing PIs, where current data is limited.

Proposed "Near-Term" PIs

- Continued effort is needed to investigate the viability several proposed PIs. It is expected that these PI's could be developed and implemented by June 1999. Actions needed to complete the development of these PIs are noted in Table 2. Other PIs associated with emergency preparedness, radiation protection, and security still require refinement and further evaluation of thresholds. PI benchmarking would be part of this activity. Use of some or all of these PIs would eliminate related baseline inspection activities.

"Long-Term" PIs

- The NRC is currently embarked on a long-term project to develop "Risk-Based" PIs (RBPI). The objective of this AEOD-sponsored program is to establish and implement a risk-based approach to PIs that is capable of evaluating trends in risk-significant performance at specific plants, and generically among groups of plants. RBPIs will use measures of reliability, availability, probability, and frequency to monitor the performance of risk-significant systems, structures, and components that contribute to core damage frequency. Quantitative data needed to support the RBPIs will come from various sources, including the industry's Equipment Performance and Information Exchange system and NRC's Reliability and Availability Data System (also under development). Industry-wide analyses of this data for common cause failures, initiating events, and system/component reliability will be used in conjunction with plant-specific analyses (Accident Sequence Precursor program and Simplified Plant Analysis Risk models) to form the basis for the RBPIs. The RBPIs may ultimately replace the indicators currently established for the Initiating Events and Mitigating System cornerstones, and should permit better, more focused use of inspection resources. This work is scheduled for completion in early 2001.

Table 2 - PROPOSED “NEAR TERM” PERFORMANCE INDICATORS

Cornerstone	Indicator	Action
Initiating Events	Shutdown safety margin (number of unplanned reductions in the safety margin for (1) reactor coolant inventory, (2) reactor coolant temperature, and (3) reactivity while the plant is shut down)	NEI proposed this indicator in their white paper; more NEI development work is needed.
Mitigation Systems	Safety System Failures	This is an NRC-developed indicator currently in use. Work is needed to establish a performance threshold before it can be added to the cornerstone framework.
	Safety system reliability (reliability of four safety systems)	NRC development in progress; more effort needed to establish meaningful thresholds.
	Shutdown operations performance (percent of outage time that defense-in-depth was compromised)	NEI plans to propose this indicator and develop the details. NRC review will also be required.
Barriers **- Reactor Coolant System**	Reactor coolant system integrity (frequency of pressure boundary leaks as defined by technical specifications, excluding steam generator tubes)	NRC plans to develop this indicator. Need to determine whether data is available, and evaluate results of benchmarking.
	Reactor coolant system integrity (percentage of individual inservice inspections that require disposition against ASME standards)	NRC plans to develop this indicator. Need to determine whether data is available, and evaluate results of benchmarking.

- The viability of PIs related to licensee implementation of the maintenance rule should be pursued. For example: (1) a PI that indicates changes in cumulative core damage frequency based on the changes resulting from on-line and shutdown safety assessments (threshold values would be plant-specific); (2) a PI that indicates structures, systems, and components that either remain in “(a)(1)” status for long periods or that have entered into “(a)(1)” status on more than one occasion over relatively short periods (threshold values would be plant-specific). Use of maintenance rule PIs would eliminate some baseline inspection activities.

- The cornerstone approach to licensee assessment relies on objective indicators of plant performance to make inferences about human reliability. However, more direct measures of pre-initiator human performance could provide a leading indication of changing plant performance. For example, errors during maintenance, testing, and operations affecting plant configurations will eventually, if not corrected, be evidenced through degraded equipment availability and reliability and increased frequency of transients and scrams. Therefore, future research may be conducted to study the utility and feasibility of developing a performance indicator(s) for pre-initiator human performance that is risk-informed and plant-performance based. In the interim, plant-level PI’s, complimented by risk-informed reviews of licensee corrective action program data, will be performed. Additionally, studies of operator error probabilities (and their contribution to plant risk) are underway. This effort is being conducted largely because this information is an essential input to the Risk-Based PIs described above.

- Review additional benchmarking data to determine if the use of dual level thresholds in the Emergency Preparedness, Public Radiation Safety, and Occupational Radiation Safety could be eliminated through consolidation.

Recommended Inspection Items

A policy decision needs to be made regarding the appropriateness of inspecting licensee assessments of Severe Accident Management Guidelines as this was a voluntary initiative by the industry.

Appendix A
Initiating Events Cornerstone

General Description

The objective of this cornerstone is to limit the frequency of those events that upset plant stability and challenge critical safety functions, during shutdown as well as power operations. When such an event occurs in conjunction with equipment and human failures, a reactor accident may occur. Licensees can therefore reduce the likelihood of a reactor accident by maintaining a low frequency of these initiating events. Such events include reactor trips due to turbine trip, loss of feedwater, loss of offsite power, and other reactor transients. There are a few key attributes of licensee performance that determine the frequency of initiating events at a plant.

Key Attributes of Licensee Performance That Contribute to Event Frequency

Those attributes of licensee performance that affect the frequency of initiating events are shown in Figure 1. They include three that were identified at the NRC's Performance Assessment Public Workshop of September 28 through October 1, 1998, (configuration control, procedure quality, and human performance) plus three additional ones (protection against external events, equipment performance, and design). Common-cause failure, which was also identified at the Workshop, has been addressed elsewhere as a cross-cutting issue. The soundness of a licensee's performance in these attributes will affect its ability (1) to maintain a low frequency of initiating events that are under the licensee's control and (2) to limit the number of initiating events caused by external factors. In the first case, the licensee can control the frequency of initiating events by ensuring adequate human performance, procedure quality, equipment performance, plant design, and configuration control. In the second case, the licensee can limit the plant's vulnerability to factors that are outside its direct control by providing adequate protection against those external factors.

Protection Against External Events
External events can cause initiating events and have been shown in some PRAs to be significant contributors to plant risk. Such events include those that are due to weather, floods, fires, accidents involving toxic substances, activities in the switchyard, grid instability, and loss of access to the ultimate heat sink. While licensees cannot prevent most of these events from occurring, they can install protective systems, such as freeze protection and lightning arresters, and implement procedures, such as shutting down prior to the arrival of a hurricane, to reduce their impact on the plant. These actions help to limit the number of plant upsets due to external events. Because external events are so rare, the lack of an initiating event due to an external event does not provide assurance that protection against such events is adequate. This attribute will be monitored by inspection of protective features.

Human Performance
Human errors can cause initiating events, especially during activities associated with plant operations, maintenance, calibration, and testing. Human-induced initiating events are relatively more frequent during shutdowns than during power operations. The nature of the work being performed while the plant is shut down is quite different from that of power operations, with more frequent, direct interactions between plant personnel and plant equipment; likewise, work scheduling is more complex because of the higher number of concurrent work activities. Hence there are more human-induced initiating events while shut down because there are more opportunities for such events. Effective planning and control of work is crucial to limiting the occurrence of human-induced initiating events, both while the plant is

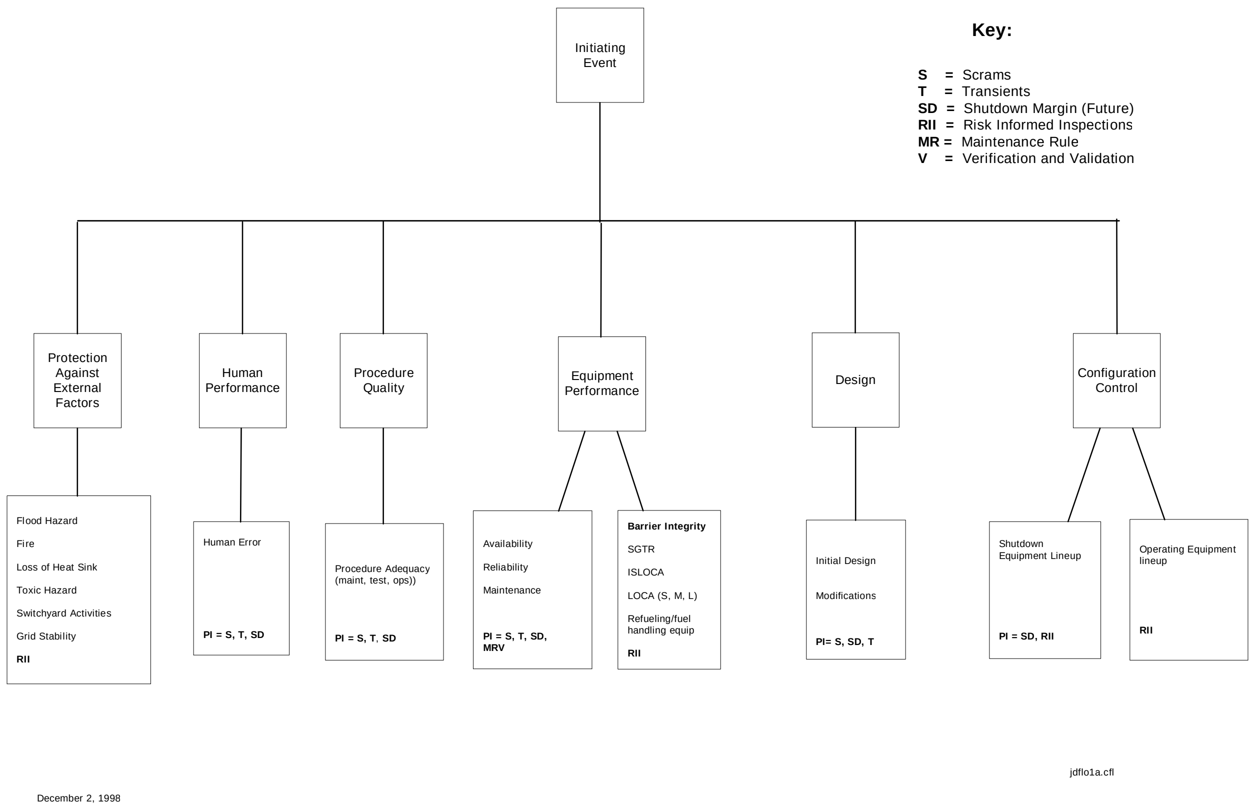


Figure 1

operating as well as when it is shut down. Human errors that cause initiating events during both shutdown and power operations will be captured by performance indicators.

Procedure Quality
Inadequate procedures can cause initiating events by inducing plant personnel to take inappropriate actions during plant operations, maintenance, calibration, or testing. This can occur for reasons such as a missing step, ambiguous or confusing language or organization, or a typographical error. Procedural inadequacies that cause initiating events will be monitored by the PIs.

Equipment Performance
Equipment failure or degradation can cause initiating events such as reactor scrams during power operations and losses of decay heat removal during shutdowns. These are expected to originate primarily in balance-of-plant (BOP) equipment while at power and in safety-related equipment during shutdowns. To limit challenges to safety functions due to equipment problems, licensees should have programs in place to achieve high availability and reliability of equipment that can cause initiating events. Strong preventive and corrective maintenance programs would be an integral part of those programs. Initiating events caused by equipment performance will be captured by PIs. In addition, licensees are required by the Maintenance Rule to establish performance criteria and goals for equipment that can cause initiating events and to monitor performance against those criteria and goals and to implement effective maintenance programs.

Barrier-related initiating events (steam generator tube rupture, loss-of-coolant-accident [LOCA], interfacing system LOCA, and fuel handling error) were judged to be unsuitable for monitoring by an indicator due their low frequency and possible high risk. Risk-informed inspections will be performed to verify that the barriers have not degraded, particularly in those areas where the safety margins are smallest.

Design
Inadequacies in either the design, the as-built configuration, or the post-installation testing of plant modifications can cause initiating events. Also, as plants age, their design bases may be misunderstood or forgotten such that an important design feature may be inadvertently removed or disabled during a plant modification. Design errors that result in initiating events will be revealed by PIs. Design errors that do not cause an initiating event are not relevant to this cornerstone.

Configuration Control
Loss of configuration control of risk-significant safety equipment (primarily support systems) can initiate a reactor transient and simultaneously compromise mitigation capability (common-cause initiators). During power operations, PIs are not viable as indicators of risk-significant configuration control problems because such events are rare and, with the extensive redundancy that exists, they would not lead immediately to a plant trip.

During shutdowns, however, when equipment is out of service for maintenance or testing, or when off-normal lineups or infrequent tests and evolutions are being conducted, configuration control problems are more likely to result in initiating events. These events will be captured by PIs (in the future) but, because of the high risk of shutdown events, PIs alone are insufficient. Risk-informed inspection of configuration control will be used to supplement the PIs during plant shutdowns.

Performance Indicators

This section defines the PIs and describes the calculational methods used to monitor licensee performance in limiting initiating events. PRAs have shown that risk is often determined by initiating events of low frequency, rather than those that occur with a relatively higher frequency. Such low-frequency, high-risk events have been considered in selecting the PIs for this cornerstone. All of the PIs used in this cornerstone are counts of either initiating events, or transients that could lead to initiating events (see Table 1). They have face validity for their intended use because they are quantifiable, have a logical relationship to safety performance expectations, are meaningful, and the data are readily available. The PIs by themselves are not necessarily related to risk. They are however, the first step in a sequence which could, in conjunction with equipment failures, human errors, and off-normal plant configurations, result in a nuclear reactor accident. They also provide indication of problems that, if uncorrected, increase the risk of an accident. In most cases, where PIs are suitable for identifying problems, they are sufficient as well, since problems that are not severe enough to cause an initiating event (and therefore result in a PI count) are of low risk significance. In those cases, no baseline inspection is required (the exception is shutdown configuration control, for which supplemental baseline inspections is necessary).

Not all aspects of licensee performance can be monitored by PIs. Risk-significant areas not covered by PIs will be assessed through inspection. Figure 1 identifies the type of monitoring (e.g., PIs or inspection) to be used for the elements of each attribute. (NEI proposed, and the Performance Assessment Workshop recommended, a PI based on the NRC's Safety System Actuations [SSA] indicator; it would only include those SSAs that occur when a plant parameter actually exceeds its set point. The Framework team is continuing to look into the use of risk-significant scrams and/or Safety System Failures (SSFs) to account for potentially high-risk initiators. Both the risk-significant scrams and the SSFs can be more closely related to risk and are therefore preferred over SSAs)

Performance Indicators for Power Operations

1. **Scrams - unplanned automatic and manual scrams while critical per 7, 000 Critical Hours[1] and risk-important scrams.**
   This measure is a count of events that upset plant stability and challenge safety functions. The indicator includes all scrams while the reactor is critical that are not directed by a normal operating or test procedure. It also includes scrams that occur during the execution of procedures in which there is a high probability of a scram but the scram was not planned. Examples of the types of scrams included are those that result from unplanned transients, equipment failures, spurious signals, human error, or those directed by abnormal, emergency, or annunciator response procedures. This is the same as the WANO indicator that is used by all U.S. plants, except that it also counts manual scrams because, from a risk perspective, they are just as important as automatic scrams. Also, a separate count is made of risk-important scrams over a 12 quarter moving sum to differentiate these scrams from the scrams without any complications. Risk-Significant Scrams = Scrams with LOCA, SGTR, LOOP, Total Loss of Heat Sink, Total Loss of Feedwater; or Scrams with a failure one or more trains of the SSPI systems. The SSPI systems are: BWRs -Emergency AC Power; High Pressure Coolant Injection Systems (HPCI, HPCS, FWCI); High Pressure Heat Removal Systems (RCIC, IC); and RHR for the suppression pool and shutdown cooling functions. PWRs -Emergency AC Power, HPSI, AFW, and RHR for the post-accident recirculation and shutdown cooling functions.
   Calculational Method - The number of scrams in the last four quarters are summed, divided by the number of critical hours in the last four quarters, then multiplied by 7, 000. This will ensure that shutdown periods are treated consistently in the PI. For risk-important scrams, the number of those scrams are added for the last 12 calendar quarters.
   Thresholds - Thresholds were determined using risk sensitivity studies as discussed in Appendix H.
   Scrams: GW - 3, WY - 6, YR - 25
   Risk-important Scrams: GW - 4, WY - 10, YR - 20
   Verification Inspection - On a sample basis, verify that the number of scrams and the critical hours are being reported accurately.

2. **Transients - unplanned changes in reactor power of greater than 20% per 7,000 Critical Hours.**
   This indicator counts unplanned events (excluding scrams) that could, in certain plant conditions, challenge safety functions. It may be a leading indicator of risk-significant events. The PI includes all changes in reactor power of greater than 20% that are not planned. It includes uncontrolled excursions in reactor power as well as unplanned controlled power reductions and shutdowns. Unplanned power reductions and shutdowns are those that are initiated before the end of the weekend following the discovery of an off-normal condition. Examples of the types of transients included are runbacks, power oscillations, power reductions conducted in response to equipment failures or personnel errors, and unplanned power reductions to perform maintenance. It does not

[1] One year of operation with an availability factor of 0.80 is equivalent to 7,000 critical hours. Rate indicators are susceptible to false positives when the denominator is small, as when a plant has been in an extended outage.

include manual or automatic scrams or load following power changes. This is similar to the information that is included by all licensees in their monthly operating reports.
Calculational Method - The number of transients in the last four quarters are summed, divided by the number of critical hours in the last four quarters, then multiplied by 7, 000. This will ensure that shutdown periods are treated consistently in the PI.
Thresholds - The threshold was determined using the industry mean plus one standard deviation based on data from July 1, 1995, through June 30, 1997. This is consistent with the methodology used by Arthur Andersen to develop the Performance Trending Tool currently being used to support the Senior Management Meeting process. Benchmarking is discussed in Appendix I.
GW - 8; others - none, not a direct measure of risk.
Verification Inspection - On a sample basis, verify that the number of transients and the critical hours are being reported accurately.

Performance Indicators for Shutdown Operations

3. **Shutdown Margin (future) - the number of unplanned decreases in the safety margins of reactor coolant level, reactor coolant temperature, and reactivity during reactor shutdown.**
This indicator counts the events that jeopardize the capability to remove decay heat from the reactor while shut down or could lead to unplanned criticality. Experience has shown that plant activities while shut down with safety equipment out of service can, under certain circumstances, have serious consequences. It is important that reactor coolant level and temperature be controlled to maintain the heat removal capability and to prevent inadvertent criticality.
Calculational Method - TBD
Thresholds - Regulatory: TBD
Safety: TBD
Verification Testing - TBD

Inspection Areas

The accuracy of the PI data reported by licensees will be verified through baseline inspections. In addition, for those elements of licensee performance that are important to risk, maintenance of defense in depth, and maintenance of safety margins and are not amenable to monitoring through PIs, licensee performance will be assessed through inspection. Table 2 identifies the type of regulatory monitoring (PIs or inspection) that will be used for the elements of each key attribute of licensee performance associated with initiating events.

Table 1 Performance Indicators for the Initiating Event Cornerstone

PI	Measured Areas	Definition	Thresholds
Scrams	Human Error, Procedure Quality, Design, and Equipment Performance	Counts unplanned automatic and manual scrams while critical; calculated per 7,000 critical hours to remove shut down periods from the indicator; Also counts risk-important scrams for a 12 quarter moving sum.	Scrams: GW - 3 WY - 6; YR - 25 Risk-important scrams - GW - 4, WY - 10, YR - 20

Transients	Human Error, Procedure Quality, Design, and Equipment Performance	Counts unplanned power excursions or controlled power reductions not included in total scrams that result in a change in reactor power of greater than 20 percent; calculated per 7,000 critical hours	GW - 8 (No others)
Shutdown Margin (future)	Human Error, Procedure Quality, Design, Equipment Performance and Equipment Lineup	Counts the number of unplanned decreases in the safety margins of reactor coolant level, reactor coolant temperature, and reactivity during reactor shutdown	TBD

Table 2 Initiating Events Key Attributes and Means to Measure

Key Attributes	**Areas to Measure**	**Means to Measure**	**Discussion**
Protection Against External Factors	All areas below	See below	Initiating events due to external factors, such as earthquakes, fires, and floods, are sufficiently rare that the absence of initiating events is no test for these protective features. Therefore, no PIs address these concerns. Risk-informed inspection will cover this area. Each area will have its own risk-informed items that are to be inspected.
	Flood Hazard	Risk-informed Inspection	site-specific
	Weather	Not Applicable	The licensee can only take mitigating actions to reduce to the effects of weather-related initiating events. Therefore, this area is covered in the mitigating systems cornerstone.
	Fire	Risk-informed Inspection	Risk-significant fires would be counted in the scrams and the operating transients indicators. As stated above, the number of those events has been small enough to preclude the use of a fire performance indicator that could provide an opportunity for early intervention. Areas for inspection for fire initiators would include review of certain important areas (control room, cable-spreading room, emergency switchgear rooms, cable vaults and tunnels etc.) for transient combustibles and elimination of ignition sources.

Key Attributes	Areas to Measure	Means to Measure	Discussion
	Loss of Heat Sink	Risk-informed Inspection	A loss of heat sink occurs when the main condenser can no longer condense steam from the power conversion system. This includes loss of heat sink not related to equipment failure, which is covered under Equipment Performance. An example would be clogging of circulating water strainers due to foreign material. An infrequent site-specific review would be conducted to verify that the potential causes of loss of heat sink that could also cause a loss of mitigating or support systems are addressed.
	Toxic Hazard	Risk-informed Inspection	site-specific
	Switchyard Activities	Risk-informed Inspection	This area was isolated from the other areas of licensee performance since these activities are typically low frequency but can have risk impact since they may result in a loss of offsite power. A review of switchyard controls would be done infrequently, focusing on those areas most likely to cause an initiating event.
	Grid Stability		Grid stability is normally excellent, but under certain conditions, such as severe weather or extended plant shutdowns, grid instability can cause initiating events at nuclear plants. The NRC is aware when such conditions exist and will follow up as required during event followup inspections. Neither PIs nor baseline inspections will monitor this area.
Human Performance	Human Error	Scrams, Transients, SD Margin (future)	Human-induced initiating events that contribute to the indicators are direct measures of initiating events for scrams and the shutdown margin. The Transient indicator measures events that may lead to initiating events. Since the transient indicator link to safety is more indirect, there will be no safety threshold for that indicator.
Procedure Quality	Procedure Adequacy (Maint., Test, Ops)	Scrams, Transients, SD Margin (future)	This factor only addresses those procedures that, if deficient, could result in an initiating event. If those procedures are inadequate such that initiating events increase, that decline in performance would be detected by the indicators since the Scram and S/D indicator are direct measures of initiating events.

Key Attributes	**Areas to Measure**	**Means to Measure**	**Discussion**
Equipment Performance	Availability, Reliability, and Maintenance	Scrams, Transients, SD Margin (future), MR V&V	Any decrease in licensee performance in this area will be manifested in increased events due to equipment performance. Like procedure quality, this is a direct measure of initiating events. This area is also monitored via the maintenance rule verification.
	Barrier Integrity	Risk-informed Inspection	Barrier-related initiating events (S/G tube rupture, LOCA, ISLOCA, & fuel handling events) were judged to be not suitable for monitoring by an indicator due the low frequency and possible high risk of those events. Inspections would be performed to verify that the barriers have not degraded, particularly in those areas where the safety margins are the smallest. This would include S/G tube ISI, risk-informed ISI reviews, and integrity of the fuel cavity used during fuel transfer.
Design	Initial Design	Scrams, Transients, SD Margin (future)	Any problems with the initial design that cause initiating events will be picked up by the indicators.
	Modifications	Scrams, Transients, SD Margin (future)	Addresses permanent and temporary modifications. Modification errors that cause initiating events would be captured by PIs.
Configuration Control	Shutdown Equipment Line-up	SD Margin (future) RII	Configuration control problems include incorrect equipment lineup, often due to frequently changing or off-normal configurations. The indicator would monitor events that cause degradation of critical safety functions during shutdown due to system configuration. Until the PI is finalized, risk-informed inspection will be performed to verify equipment line-ups, particularly during special tests or evolutions.
	Operating Equipment Line-ups	Risk-informed Inspection	Configuration control problems can cause a trip and the simultaneous loss of a mitigating system or function (common-cause initiating events). A PI is not viable because such events are rare and, with extensive redundancy during operation, they would not lead immediately to a plant trip. Inspection would also be focused on emergent work items where less time was available for the licensee to plan the work.

Appendix B
Mitigating Systems Cornerstone

General Description

The objective of this cornerstone is to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). When such an event occurs in conjunction with equipment and human failures, a reactor accident may result. Licensees therefore reduce the likelihood of reactor accidents by enhancing the availability and reliability of mitigating systems. Mitigating systems include those systems associated with safety injection, residual heat removal, and emergency AC power. This cornerstone includes mitigating systems that respond to both operating and shutdown events. There are several key attributes of licensee performance that ensure adequate mitigating system performance at nuclear power plants.

Key Attributes of Licensee Performance That Contribute to Mitigating Systems Performance

Those attributes of licensee performance that are important to mitigating system performance at a plant are: protection against external events, design, configuration control, equipment performance, procedure quality, and human performance. These attributes embrace and refine the key attributes described in a report entitled, "Results of the NRC's Performance Assessment Public Workshop," (LANL), October 25, 1998) and are shown in Figure 1. The quality of these attributes will affect the licensee's ability to optimize the availability and reliability of the mitigating system function. The licensee can ensure mitigating system performance by supporting effective human performance, procedure quality, equipment performance, plant design, and configuration control.

For each of these attributes, specific elements have been identified. For example, the particular aspects of configuration control to ensure adequate mitigating system performance include equipment lineup during operating and shutdown modes, control of temporary modifications and operator work-arounds, and risk-informed equipment maintenance scheduling. As another example, the types of procedures that are relevant to mitigating system performance include maintenance, test, and operating procedures. The discussions that follow summarize the relationship between the key attributes and mitigating system performance.

Protection Against External Events
External events can prevent mitigating systems from performing their intended functions by reducing their capability or rendering the systems inoperable. Most of these factors are site-specific, related to weather, loss of heat sink, toxic and flood hazards, and seismic hazards. Fire can also prevent mitigating systems from functioning, and was included in this area because fire is typically analyzed in the IPEEE. Due to the rare but possibly risk-important nature of these events, no PI was judged suitable to monitor licensee performance in this area. Risk-informed inspection will be performed in these areas.

Design
Inadequacies in the initial design or in the control of plant modifications can affect the capability of mitigating systems to perform their intended function as well as their availability and reliability. As plants age, their design bases may be lost such that an important design feature may be inadvertently

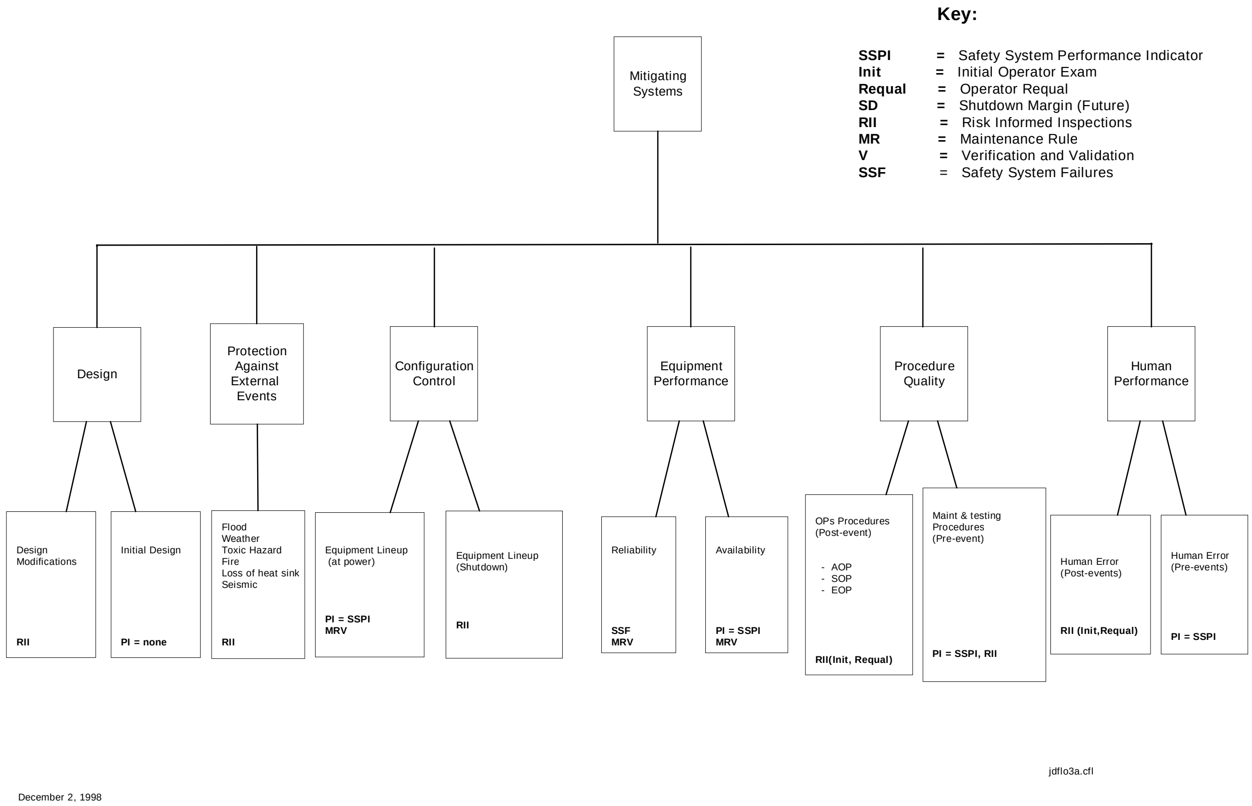


Figure 1

altered or disabled. It is expected that PIs can be used to provide partial information regarding the adequacy of the initial design and the control of design modifications insofar as they monitor the availability and reliability of equipment. Many aspects of the original plant design have been adequately addressed by initial design reviews, start-up testing, 50.54(f) reviews, and periodic surveillance programs. Inspection in this area should be limited to those risk-significant design features and design assumptions, if any, not adequately addressed in previous programs. In addition, periodic design basis reviews using plant modifications as a window into the original design, can help maintain confidence that mitigating systems will respond to events as intended. The modifications reviewed should be only those modifications that could alter the functionality of mitigating systems used during risk-significant accident sequences. Also, risk-informed inspection of those areas that could affect the functionality of mitigating systems is warranted to insure that the design and design basis was not inadvertently altered. PIs are not expected to address the understanding and control of conditions outside the design basis.

Configuration Control
Loss of configuration control of risk-significant safety equipment (primarily support systems) can compromise mitigation capability. When safety systems are not available or system redundancy is degraded due to misaligned valves or switches, that unplanned unavailability will be captured by the PIs for selected systems. For other systems not covered by the PIs, risk-informed verification of systems and components in a standby status is planned for both the operating and shutdown conditions. Also, the maintenance rule and the associated verification will also monitor operating performance in this area.

Equipment Performance

Adequate availability and reliability of equipment important to effective performance of mitigating systems is critical to mitigating the impact of initiating events on plant safety. The performance of certain mitigating systems are measured by the PIs to the extent that testing is adequate to measure functional availability and reliability. In addition, the performance of all structures, systems, and components (SSCs) important to the performance of mitigating systems will be monitored as part of licensees' implementation of the maintenance rule. Consequently, performance indicator data will be supplemented by verification of maintenance rule implementation.

Procedure Quality
To ensure proper functioning of mitigating systems, the procedures which control their maintenance and testing operation must be correct. Maintenance and testing procedures influence the capability of mitigating systems to respond to initiating events. The quality of such procedures are indirectly confirmed by the performance of mitigating systems as monitored by the PIs and verification inspection of maintenance rule implementation. Test procedures will be reviewed to identify what post-accident mission-related aspects of the design are not tested. This would be an input to the design inspection.

Emergency and abnormal operating procedures are also essential for mitigating system performance. Initial and requalification testing of operators provides an indication of the quality of operating procedures, including abnormal operating procedures, standard operating procedures, and emergency operating procedures.

Human Performance
Human performance in day-to-day, pre-initiator plant activities influences the performance of mitigating systems through the conduct of maintenance and test activities. Therefore, the licensee's problem identification and resolution program is expected to identify and correct human errors that lead to degraded plant performance which is measured by other plant performance indicators for mitigating systems, including those associated with design, configuration control, and equipment performance. Also, human errors that degrade equipment will be monitored through maintenance rule implementation.

Human actions are also clearly important in plant response to initiating events. Further, human performance is critical to mitigation in multiple-failure accident sequences. Examples of human actions that are important to the performance of mitigating systems are those associated with depressurization and cool down and actions involved in aligning and recovering backup cooling water systems. While few data are available to directly measure post-initiator human performance, operator performance during initial and requalification examinations provide an indirect indication of expected post-initiator operator performance.

Performance Indicators

This section defines the PIs used to monitor licensee performance in mitigating the effects of initiating events, describes their calculational methods and thresholds, and identifies the inspections necessary to verify their accuracy (see Table 1). While safety systems and components are generally thought of as those that are designed for design-basis accidents, not all mitigating systems have the same risk importance. PRAs have shown that risk is often influenced not only by front-line mitigating systems, but also by support systems and equipment. Such systems and equipment, both safety- and nonsafety-related, have been considered in selecting the PIs for this cornerstone. The PIs are all direct counts of either mitigating system availability or reliability or surrogates of mitigating system performance. They have face validity for their intended use because they are quantifiable, have a logical relationship to safety performance expectations, are meaningful, and the data are readily available. Not all aspects of licensee performance can be monitored by PIs. Risk-significant areas not covered by PIs will be assessed through inspection. Figure 1 identifies the type of monitoring (i.e., PIs or inspection) to be used for the elements of each attribute.

Performance Indicators for Power Operations

1. **Safety System Performance Indicator (SSPI) - the INPO indicator of the performance of four of the most risk-significant safety systems.** This indicator monitors several generic risk-significant safety systems. The SSPI systems for BWRs include high-pressure injection systems (high-pressure coolant injection or high-pressure core spray or feedwater coolant injection), high pressure heat removal systems (reactor core isolation cooling or isolation condenser), residual heat removal systems, and emergency AC power systems. For PWRs, the systems monitored include high-pressure safety injection systems, auxiliary feedwater systems, residual heat removal systems, and emergency AC power systems.

   The SSPI indicator provides a limited but useful sample of safety system performance information associated with equipment important to risk. Limitations in scope of the SSPI are augmented by review of implementation of the maintenance rule on those systems not covered by the SSPI, with focus on issues that cross cornerstones such as common cause failure and human performance.

   a. <u>SSPI Unavailability</u>. This indicator measures the in-service unavailability of four generic risk-significant safety systems. The SSPI for each monitored system is the average of the unavailability of the individual trains that comprise the system.

   Calculation Method - The SSPI for each monitored system is the average of the unavailabilities of the individual trains that comprise the system. Each train unavailability is the ratio of its unavailable hours to the hours the system was required to be operable. The train unavailable hours is the sum of the planned, unplanned, and fault exposure unavailable hours. Detailed definitions of these terms are contained in INPO 96-003.

   Thresholds - The following thresholds were determined following a sensitivity analysis of risk information as discussed in Appendix H.

   HPCI: GW - 0.04; WY - 0.12; YR - 0.5
   RCIC: GW - 0.04; WY - 0.12; YR - 0.5
   HPCS: GW - 0.015; WY - 0.04; YR - 0.2
   Emergency Power System: GW - 0.025; WY - 0.05 (0.1 >2 EDG);
   YR - 0.1 (0.2 >2EDG)[2]
   BWR RHR: GW - 0.015; WY - 0.05; YR - TBD
   AFW: GW - 0.02; WY - 0.06; YR - 0.12
   PWR Hi Pressure Injection: GW - 0.015; WY - .05; YR - TBD

   Verification Inspection - Selected review of a sample of the SSPI systems to verify that unavailability data are reported accurately

   b. SSPI Unreliability (future) - This indicator measures the demand unreliability of the above described generic risk-significant safety systems to start and/or operate for the prescribed period of

[2]Oconee thresholds are TBD since they do not have emergency diesel generators.

time to perform a safety function. The SSPI for each monitored system is the average of the unreliability of the individual trains that comprise the system. Each train unreliability is the ratio of the number of start or run failures to the number of demands or run hours respectively. Current data showed wide fluctuations and needs further review to establish as a PI. Detailed definitions of these terms and prescriptions for combining failures are contained in INPO 96-003.

Calculation Method - TBD

Thresholds - TBD

Verification Inspection - Selected review of a sample of the SSPI systems to verify that demand and failure data are reported accurately.

2. **Safety Systems Failures - events or conditions that could prevent the fulfillment of the safety function of structures, systems, or components.**
This measure is a count of the number of events or conditions that did prevent, or could have prevented, the fulfillment of the safety function of any of 26 safety-related structures, systems, and components. For systems consisting of multiple redundant trains, failure of all trains is necessary for a safety system failure. The indicator also counts failures that cause at least one independent train or channel to become inoperable in multiple systems. This is the same indicator used in the NRC Performance Indicator program. We recognize that this indicator measures more than mitigating systems. However, this indicator was still added in mitigating systems since the SSPI reliability indicator could not be used without further analysis.

Calculational Method - The number of safety system failures in the last four quarters are summed (a four-quarter moving sum).

Thresholds - The threshold was determined using the industry mean plus one standard deviation based on data from July 1, 1995, through June 30, 1997. This is consistent with the methodology used by Arthur Andersen to develop the Performance Trending Tool currently being used to support the Senior Management Meeting process. Benchmarking is discussed in Appendix I.
GY: 5
Others: None

Verification Inspection - On a sample basis, verify that the number of safety system failures are being reported accurately.

Performance Indicator for Shutdown Operations

3. **PI for Shutdown Operations (future) - mitigating system availability during shutdown.** Most licensees manage shutdown risk in accordance with NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management." They manage defense in depth, through configuration control, for key safety functions (decay heat removal, inventory control, electrical power availability, reactivity control and containment). This PI will measure the percent of outage time that each key safety function lacked defense in depth, either from installed equipment or contingency actions. Since defense in depth for each area would need to be defined and further additional work is needed, this PI will be developed in the future.

Calculation Method - TBD
Thresholds - TBD
Verification Inspection - TBD

Inspection Areas for Mitigating Systems

The accuracy of the PI information reported by licensees will be verified through baseline inspections. In addition, for those elements of licensee performance that are important to risk, maintenance of defense in depth, and maintenance of safety margins and are not amenable to monitoring through PIs, licensee performance will be assessed through inspection. Table 2 identifies the type of regulatory monitoring (i.e., PIs or inspection) which will be used for the elements of each key attribute of licensee performance associated with mitigating systems.

Table 1. Performance Indicators for the Mitigating Systems Cornerstone

PI	Measured Area	Definition	Thresholds
SSPI Availability	availability of specified risk-important mitigating systems	For each monitored system, counts the average of the unavailabilities of the individual trains that comprise the system. Each train unavailability is the ratio of its unavailable hours to the hours the system was required to be available. The train unavailable hours is the sum of the planned, unplanned, and fault exposure unavailable hours.	HPCI & RCIC : GW - 0.04; WY - 0.12; YR - 0.5 HPCS: GW - 0.015; WY - 0.04; YR - 0.2 EDGs: GW - 0.025; WY - 0.05 (0.1 >2EDG); YR - 0.1 (0.2 >2 EDG)[3] BWR RHR: GW - 0.015; WY - 0.05; YR - TBD AFW: GW - 0.02; WY - 0.06; YR - 0.12 PWR Hi Pressure Injection: GW - 0.015; WY - .05; YR - TBD
SSPI Reliability **(future)**	reliability of mitigating systems	For each of four monitored systems, calculates the demand unreliability to start and/or operate for the prescribed period of time to perform their safety functions.	TBD
Safety System Failures	reliability of mitigating systems	For each of 26 safety-related structures, systems, and components, counts the number of events or conditions that did prevent, or could have prevented, the fulfillment of the safety function as a four quarter moving sum.	GW - 5; others - none.
Mitigating system availability during shutdown **(future)**	availability of mitigating systems to limit shutdown risk.	Plan to calculate the outage time that each key safety function lacked defense in depth, either from installed equipment or contingency actions. Since defense in depth for each area would need to be defined, this PI will be developed in the future.	TBD

[3]Oconee thresholds are TBD since they do not have emergency diesel generators.

Table 2. Mitigating Systems Key Attributes and Means to Measure

Key Attributes	Areas to Measure	Means to Measure	Comment
Protection against external factors	All external factors listed below.	Risk-informed (R-I) inspection	External factors can prevent mitigating systems from responding if called upon. Since the systems that mitigate external events are called upon so rarely, inspection (as warranted) of mitigating systems and design modifications will verify that the systems remain in place and are functional. Inspections of this key attribute will be very plant specific. Some plant features that are important to risk are discussed below.
	Flood	R-I inspection	Protection against the effects of floods is afforded in a variety of ways that includes drains, encasing equipment in splash-proof barriers, and providing barriers, such as flood doors between redundant trains of systems. These protection systems are site-specific and should be subjected to an inspection that is commensurate with the risk importance of the feature and system.
	Weather	R-I inspection	In general, most safety systems are well protected against the effects of weather by being enclosed in protective structures. However, there are certain portions of systems that are susceptible to effects of weather for which protection is provided by design. Examples include: fluid lines outside buildings that could freeze may be protected by lagging or trace heating, ventilation intakes and roof drains that could become blocked are protected by covers or grilles. The inspection will review those design features used to protect multiple mitigating systems from the effects of weather, including potential common cause effects on mitigating systems.
	Toxic hazard	R-I inspection	Plant-specific.
	Fire	R-I inspection	System functions are typically protected against fires by providing protection by fire barriers with or without detection and suppression, and by establishing barriers between different trains of redundant systems. The status of passive and active fire protection measures should be inspected in a risk-informed way. The first areas for inspection would include the functionality of detection and suppression systems (including the fire brigade) located in important areas (e.g., control room, cable-spreading room, emergency switchgear rooms, cable vaults and tunnels). Less frequently, inspection would be performed to verify that if the fire was not extinguished (i.e., defense in depth is ineffective), the fire would not spread (i.e., fire barriers are intact for those areas) and important alternate actions and stations are available to safely shutdown the plant.

Key Attributes	Areas to Measure	Means to Measure	Comment
	Seismic	R-I inspection	Safety-significant equipment is designed for seismic events by being seismically qualified and having appropriate anchorage. Since this is unlikely to change and has been reviewed industry-wide (A-46, masonry walls, anchor bolts, pipe supports), the focus of the inspection would be to ensure that plant modifications (e.g., installation of scaffolding or removal of snubbers) have not compromised the capability of mitigating systems during seismic events and that the qualification of equipment was maintained to prevent the introduction of a new common cause failure.
	Loss of heat sink	R-I inspection	The ultimate heat sink for systems that provide cooling for the front-line and support systems is typically the same source as the circulating water, although in some plants there is a dedicated supply. In either case, they are susceptible to the same external effects as circulating water, such as clogging of strainers due to foreign material. Site-specific inspection will assess whether the required features to prevent loss of supply are not degraded. This inspection ought to focus on the potential common-cause failures of the heat sink, most notably service water heat exchanger fouling.
Design	Initial design	None	PIs would only indicate problems with the initial design after the mitigating systems are called upon to act, which would be too late. Initial design reviews have been extensive the last several years, particularly in response to the 50.54(f) letters. Further inspection of initial design would be performed in those areas where plant modifications have been made. In addition, risk-informed inspection will review those aspects of the design not subject to periodic testing, to assure that those features are still functional.
	Design modifications	R-I inspection of design modifications	As above, PIs would not provide timely indication of faulty plant modifications. The focus of inspection in this area is to ensure that risk-significant mitigating systems remain functional after modifications, both intentional and inadvertent. That inspection ought to focus on the design interfaces, configuration management, post-modification testing, and those areas not readily verified by testing (EQ, seismic, etc.) that are risk significant.
Configuration Control	Equipment Line-up (at power)	SSPI, MRV	For those systems monitored, SSPI will provide some information on the adequacy of configuration control, especially on licensee programs and practices to maintain critical safety functions with adequate margins. Inspections will monitor plant configurations that affect mitigating system performance, especially for system restoration, as part of maintenance rule verification [i.e., A(4)].

Key Attributes	Areas to Measure	Means to Measure	Comment
	Equipment Line-up (shutdown)	R-I inspection	A future PI may measure events that cause degradation of critical safety function during plant shutdown based on mitigating system configuration. In the interim, inspection will be conducted of the licensee's program to manage shutdown risk.
Equipment Performance	Availability	SSPI, MRV	The SSPI will monitor the unavailability of certain important systems and licensees also monitor the availability of the SSCs of mitigating systems as part of Maintenance Rule implementation.
	Reliability	SSFs, MRV	Licensees monitor the reliability of the SSCs of mitigating systems as part of Maintenance Rule implementation. SSFs are an interim indicator until the SSPI Reliability indicator is developed. The MR review could be eliminated for those monitored systems when the SSPI Reliability indicator is ready.
Procedure Quality	Pre-event maintenance & test procedures	SSPI, R-I inspection	Equipment performance (e.g., as monitored through maintenance rule implementation) and the SSPI will indirectly confirm the quality of maintenance and test procedures. Test procedures will be reviewed to identify what post-accident mission-related aspects of the design are not tested. This would be an input to the design inspection.
	Post-event operating procedures	Initial operator exams & requalification program inspections	These procedures are not used until after an event occurs, thus a PI is not suitable to measure the quality of these procedures. Review of emergency and abnormal operating procedures, most likely during initial and requalification testing of operators, provides some confirmation of the quality of mitigating system operating procedures on a sample basis. Inspection, as part of the review of design modifications, may also will identify procedure inadequacy.
Human Performance	Pre-event human errors	SSPI	Pre-event errors will be monitored by the SSPI since errors in the operating and maintaining the equipment will be reflected in system unavailability. Also, when mitigating system equipment performance is degraded, then the role of human performance is expected to be assessed by the licensee as part of its problem identification and resolution program.
	Post-event human errors	Initial operator exams & requalification program inspections	Current PIs will not provide indication of post-event human performance. Operator performance during initial and requalification examinations provide an indication of post-event operator performance.

Appendix C
Barrier Integrity Cornerstone

C.1 General Description

The purpose of this cornerstone is to provide reasonable assurance that the physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. These barriers play an important role in supporting the NRC Strategic Plan goal for nuclear reactor safety, "Prevent radiation-related deaths or illnesses due to civilian nuclear reactors." The defense in depth provided by the physical design barriers which comprise this cornerstone allow achievement of the reactor safety goal.

The first barrier is the fuel cladding. Maintaining the integrity of this barrier prevents the release of radioactive fission products to the reactor coolant system, the second barrier. Maintaining the integrity of the reactor coolant system reduces the likelihood of loss of coolant accident initiating events and prevents the release of radioactive fission products to the containment atmosphere in transients and other events. Even if significant quantities of radionuclides are released into the containment atmosphere, maintaining the integrity of the third barrier, the containment, will limit radioactive releases to the environment and limit the threat to the public health and safety. Therefore, there are three desired results associated with the barrier integrity cornerstone. These are to maintain the functionality of the fuel cladding, the reactor coolant system, and the containment.

For this discussion, the scope of the fuel cladding barrier includes the fuel cladding during operations, shutdown, and refueling, both inside containment and in the spent fuel pool. The scope of the reactor coolant system barrier includes piping and pressure retaining components such as valves, pumps, seals, and gaskets. It also includes portions of connected systems when the plant configuration is such that these connected systems form a part of the reactor coolant system pressure barrier. Although steam generator tubes are a part of the barrier, they are being addressed under the initiating events cornerstone. The scope of the structures, systems, and components related to the containment barrier includes the primary and secondary containment buildings (including personnel airlocks and equipment hatches), primary containment penetrations and associated isolation systems, and risk-significant systems and components necessary for containment heat removal, pressure control, and degraded core hydrogen control.

C.2 Key Attributes of Licensee Performance that Contribute to Barrier Integrity

The concept of the cornerstone approach, including the barrier integrity cornerstone, was discussed in the Performance Assessment Workshop held in Bethesda, MD, on September 28 through October 1, 1998. During a breakout session for further development of the barrier integrity cornerstone, the working group expanded its specific focus from containment systems to barriers. After extended consideration, the workshop attendees determined the barriers should be subdivided into three categories: fuel cladding, reactor coolant system, and containment. In order to achieve the desired results, the group then determined that the key attributes of these three elements should be: (1) Design Control, (2) Human Performance, (3) Procedure Quality, (4) Configuration Control, and (5) Equipment/Barrier Performance. The NRC staff determined that these were the appropriate key attributes for further development.

Specific areas to measure were identified for each of the noted key attributes. The means to measure performance in each of these specific areas were also identified. These means include the use of performance indicators, risk-informed inspection activities, and licensee corrective action programs. The following sections discuss each of the key attributes, the areas to measure, and the recommended performance indicators and risk-informed inspections and oversight activities needed to support each of the three barriers comprising the overall barrier integrity cornerstone. Diagrams depicting the barrier integrity cornerstone, along with the key attributes and the areas and means of measurement, are shown in Figures C1, C2, and C3. Table C1 is a summary of the proposed performance indicators associated with the barrier integrity cornerstone. Table C2 is a summary table for the barrier integrity cornerstone which provides further information on the means of measurement.

C.3 Key Attributes Affecting Fuel Cladding

C.3.1 Design Control

Licensees are responsible for the oversight of nuclear fuel vendors regarding their design and manufacturing quality of the actual fuel pins. Vendor quality assurance programs and oversight should detect errors with regard to manufacturing, packaging, shipping, etc. Because of this, reactor licensees need not be individually inspected or assessed with regard to nuclear fuel design quality. Undetected fuel pin or assembly manufacturing errors should be revealed during startup physics testing. If significant problems were detected, shutdown would be accomplished and corrective actions taken, avoiding significant risk.

Proper reactor core design is essential to assuring that subsequent power operation can be conducted without challenging the integrity of the fuel cladding. The core design analysis, including the core operating limits report and the reload analysis, establishes the operational limitations for core power operation, with sufficient margin to ensure that thermal limits are not exceeded during anticipated transients. Core design analyses must be completed with sufficient rigor and quality to demonstrate that, in the proposed core configuration, the nuclear fuel cladding will maintain its integrity.

The conduct of physics testing during startup following refueling activities in part provides a verification that the reactor core exhibits the characteristics predicted by the design analysis. This testing is conducted prior to any significant power operation so that errors during testing would not be likely to cause any fuel cladding degradation. The proper completion of physics testing is essential to ensure that the core design will adequately support subsequent high power reactor operation without challenging the established thermal limits and ultimately the nuclear fuel cladding.

The reactor coolant system activity performance indicator may be used as a means of measuring performance in this key attribute.

C.3.2 Human Performance

Nuclear fuel cladding integrity can be challenged by inappropriate human actions, including improperly performed reactivity manipulations, inadequate chemistry control practices, improper implementation of foreign material exclusion programs, and inappropriately positioned fuel assemblies during refueling, as examples. The introduction of foreign material into the reactor vessel or connected systems could lead to degraded fuel barrier performance by limiting coolant flow past fuel pins or assemblies or by damaging fuel cladding as a result of direct impact on fuel cladding surfaces. Foreign material could also cause mitigating systems such as control rods to fail or be degraded.

The RCS activity performance indicator may be used as a measure of performance in this key attribute. Licensee problem identification and corrective action programs should provide adequate assurance that adverse trends in human performance, particularly as they relate to the barrier integrity cornerstone, are promptly identified and corrected.

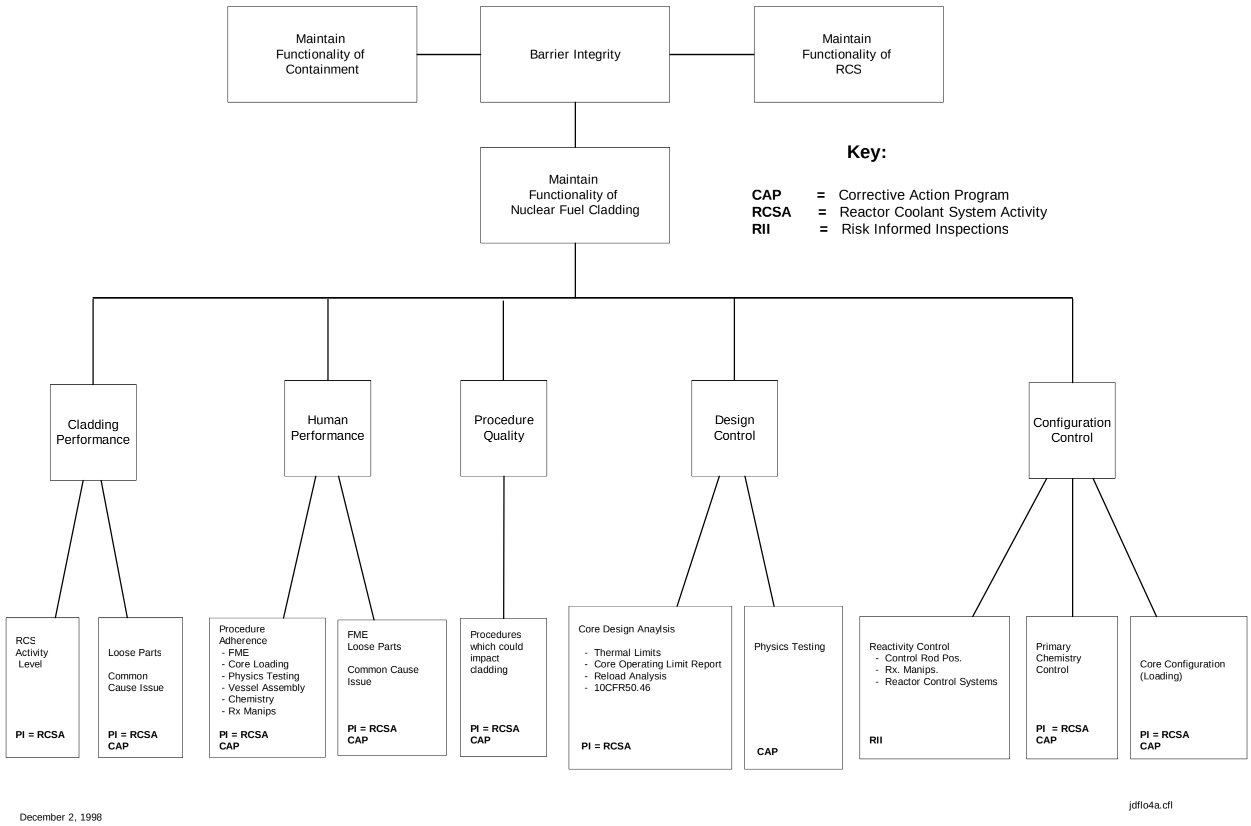

Maintain Functionality of Containment
Barrier Integrity
Maintain Functionality of RCS
Key:
CAP = Corrective Action Program
RCSA = Reactor Coolant System Activity
RII = Risk Informed Inspections
Maintain Functionality of Nuclear Fuel Cladding
Cladding Performance
Human Performance
Procedure Quality
Design Control
Configuration Control
RCS Activity Level
PI = RCSA
Loose Parts
Common Cause Issue
PI = RCSA
CAP
Procedure Adherence
- FME
- Core Loading
- Physics Testing
- Vessel Assembly
- Chemistry
- Rx Manips
PI = RCSA
CAP
FME
Loose Parts
Common Cause Issue
PI = RCSA
CAP
Procedures which could impact cladding
PI = RCSA
CAP
Core Design Analysis
- Thermal Limits
- Core Operating Limit Report
- Reload Analysis
- 10CFR50.46
PI = RCSA
Physics Testing
CAP
Reactivity Control
- Control Rod Pos.
- Rx. Manips.
- Reactor Control Systems
RII
Primary Chemistry Control
PI = RCSA
CAP
Core Configuration (Loading)
PI = RCSA
CAP
December 2, 1998
jdflo4a.cfl

Figure C1

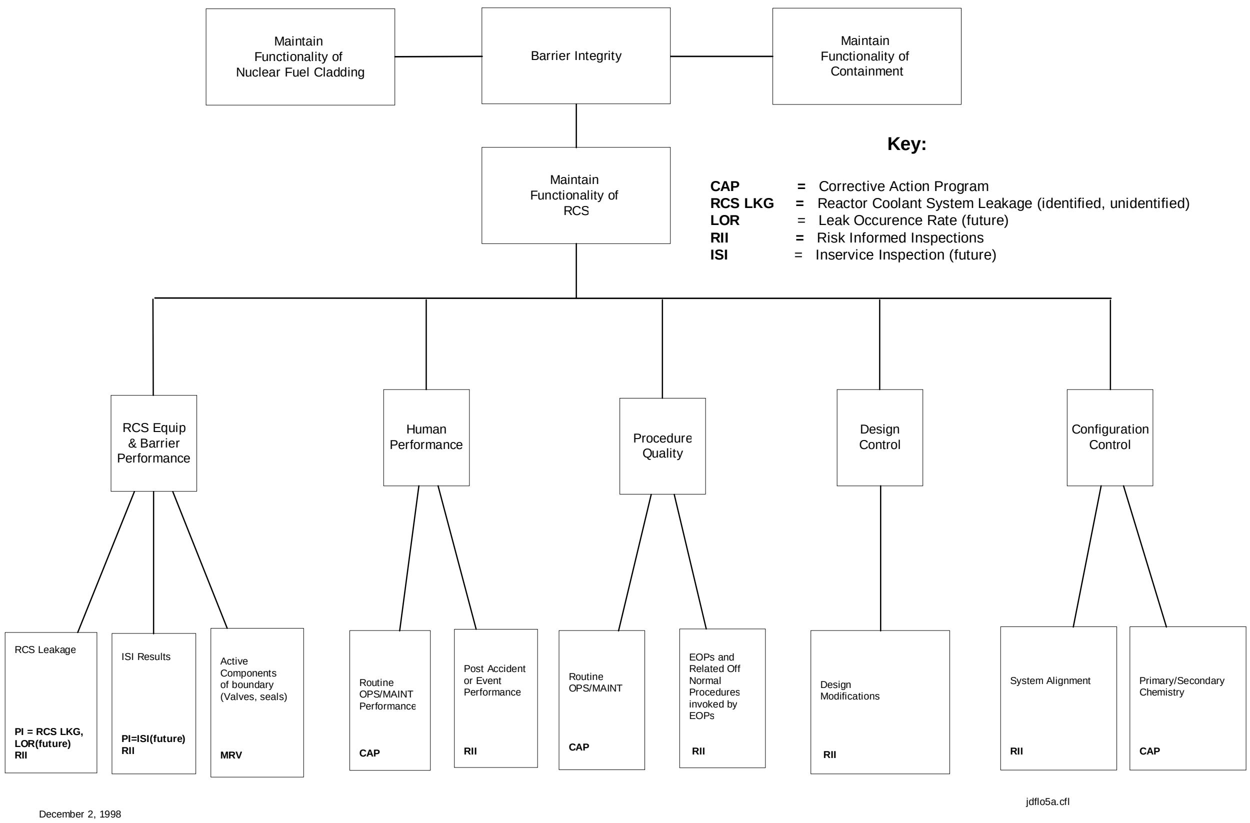
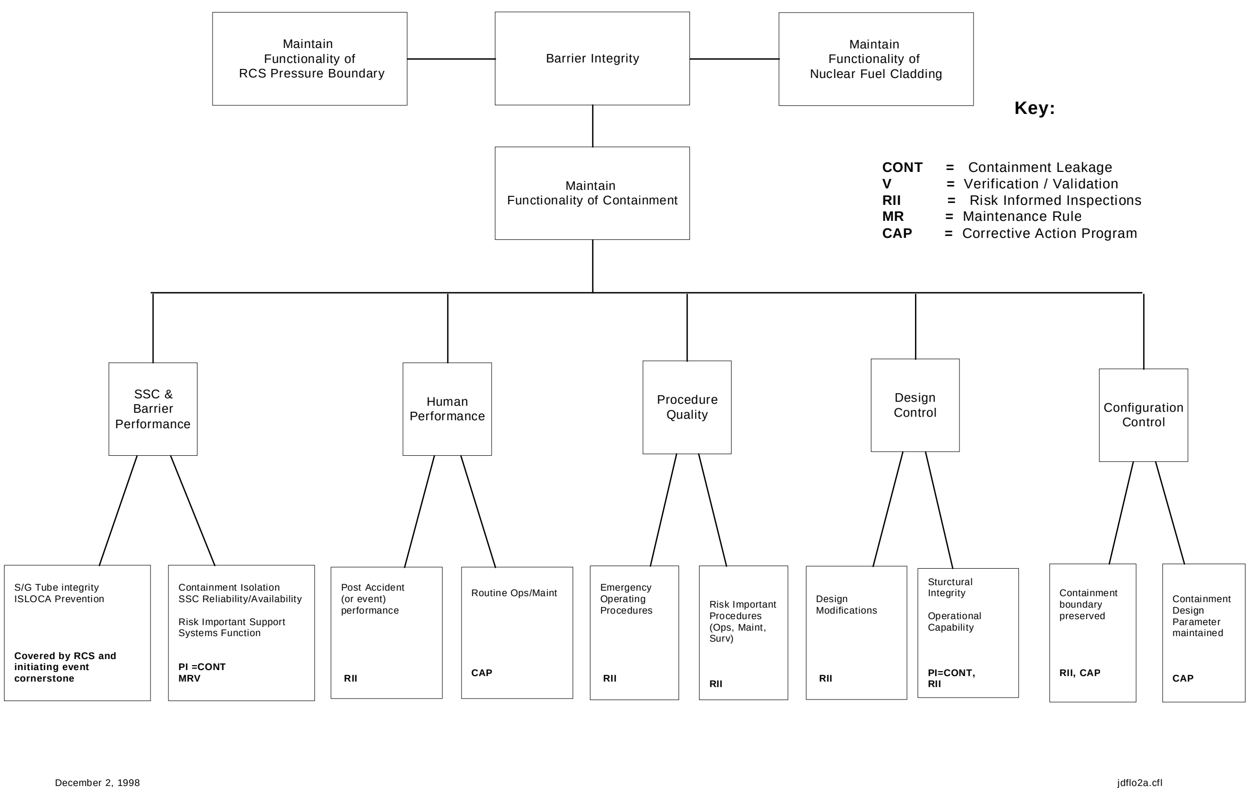

Maintain Functionality of Nuclear Fuel Cladding
Barrier Integrity
Maintain Functionality of Containment
Maintain Functionality of RCS
Key:
CAP = Corrective Action Program
RCS LKG = Reactor Coolant System Leakage (identified, unidentified)
LOR = Leak Occurence Rate (future)
RII = Risk Informed Inspections
ISI = Inservice Inspection (future)
RCS Equip & Barrier Performance
Human Performance
Procedure Quality
Design Control
Configuration Control
RCS Leakage
PI = RCS LKG, LOR(future)
RII
ISI Results
PI=ISI(future)
RII
Active Components of boundary (Valves, seals)
MRV
Routine OPS/MAINT Performance
CAP
Post Accident or Event Performance
RII
Routine OPS/MAINT
CAP
EOPs and Related Off Normal Procedures invoked by EOPs
RII
Design Modifications
RII
System Alignment
RII
Primary/Secondary Chemistry
CAP
December 2, 1998
jdflo5a.cfl

Figure C2



Maintain Functionality of RCS Pressure Boundary
Barrier Integrity
Maintain Functionality of Nuclear Fuel Cladding
Key:
CONT = Containment Leakage
V = Verification / Validation
RII = Risk Informed Inspections
MR = Maintenance Rule
CAP = Corrective Action Program
Maintain Functionality of Containment
SSC & Barrier Performance
Human Performance
Procedure Quality
Design Control
Configuration Control
S/G Tube integrity
ISLOCA Prevention
Covered by RCS and initiating event cornerstone
Containment Isolation
SSC Reliability/Availability
Risk Important Support Systems Function
PI =CONT
MRV
Post Accident (or event) performance
RII
Routine Ops/Maint
CAP
Emergency Operating Procedures
RII
Risk Important Procedures (Ops, Maint, Surv)
RII
Design Modifications
RII
Sturctural Integrity
Operational Capability
PI=CONT,
RII
Containment boundary preserved
RII, CAP
Containment Design Parameter maintained
CAP
December 2, 1998
jdflo2a.cfl

Figure C3

C.3.3 Procedure Quality

Procedures that direct activities which have the potential to affect fuel cladding integrity must be adequately established and maintained. Procedures included in this area involve reactivity control, foreign material exclusion, chemistry control, refueling activities, fuel handling, reactor vessel assembly, and physics testing. Inadequate procedures could cause problems which lead to degradation of fuel cladding integrity.

The reactor coolant system activity performance indicator may be used as a means of measuring performance in this key attribute. To the extent that there are procedure deficiencies associated with the above noted activities, they should be identified as root causes of problems in other areas, including the configuration control key attribute. Additionally, adverse trends involving procedure deficiencies should be resolved by effective implementation of individual licensee corrective action programs.

C.3.4 Configuration Control

Fuel cladding degradation can result from either inadequate human or equipment performance. With regard to human performance, refueling operators must ensure that new and previously-used nuclear fuel assemblies are properly handled and stored, properly positioned, and correctly oriented in the specified core locations. Control rod positions (patterns) during plant operation must be properly established and maintained. Plant operators must conduct reactivity manipulations in a well-controlled and deliberate manner. With regard to equipment performance, reactivity control systems (including control rod drives) must be properly configured and maintained. Some baseline inspection is warranted in this area because of the risk to fuel cladding integrity associated with either inadequate human or equipment performance in the above described areas.

Maintaining proper water chemistry in the reactor coolant system (RCS) is also essential to the long term reliability of both the nuclear fuel and the RCS pressure boundary. A failure to maintain the proper chemistry conditions has the potential to result in degradation (and ultimately failure) of the nuclear fuel cladding**.** The reactor coolant system activity performance indicator may be used as a means of measuring performance in this key attribute. Additionally, adverse trends involving configuration control should be resolved by effective implementation of individual licensee corrective action programs.
.

C.3.5 Equipment /Barrier Performance

Though it would be preferable to assess the extent of fuel cladding degradation rather than monitor actual cladding failures, a practical means of conducting such an assessment is not available. As a result, a means of monitoring fuel cladding failures must be established. Since perforation of nuclear fuel cladding results in the release of fission products to the RCS, increases in RCS radioactivity level can be directly correlated to the integrity of the fuel cladding barrier. A performance indicator which trends RCS radioactivity level provides an objective means of assessing the overall performance of the nuclear fuel cladding. In boiling water reactors, fuel cladding failures will also be detected by main steam line or condenser offgas radiation monitors.

Loose parts in the reactor coolant system, most importantly in the reactor vessel, can lead to various problems, including damage to the nuclear fuel cladding, either by direct impact on the fuel pins or by limiting RCS fluid flow past individual pins or assemblies. Minimizing the number of frequency of loose parts in the reactor vessel is partly controlled by licensee foreign material exclusion (FME) programs, however loose parts can also be introduced by degradation or failures of components which are internal to the reactor coolant or connected systems. Monitoring and limiting the frequency of reactor vessel loose part events should reduce the potential for fuel cladding failures.

C.4 Key Attributes Affecting Reactor Coolant System

C.4.1 Design Control

Maintaining confidence in loss of coolant accident frequency estimates requires assuring the quality of design modification activities which can potentially impact the RCS strength margins and therefore the likelihood of an RCS pressure boundary rupture. This assumes that the original design of the RCS was adequate and has been proven through hydrostatic testing. The quality of RCS design modification implementation will be measured in a risk-informed manner through specific requirements in the baseline inspection program. No performance indicator has been developed to measure the design control attribute.

C.4.2 Human Performance

Human performance can affect RCS integrity through routine operation and emergency operation and through maintenance and surveillance activities. Proper performance of these activities helps maintain assurance that LOCA frequency does not increase significantly.

Operator errors which cause RCS heatup or cooldown or pressure/temperature limits to be exceeded provide a leading indication of the potential for future pressure boundary leaks. Such events can cause existing microscopic cracks in passive RCS pressure boundary components to grow. Verification that the results of licensee engineering analysis following an event of either excessive heatup or cooldown, or operation outside of allowable pressure/temperature limits, are satisfactory and that related human factors have been corrected will be performed on a reactive basis as needed and will not be a part of the baseline inspection program.

Most human performance deficiencies related to routine maintenance and surveillance testing of the RCS have not been shown to be particularly risk significant and will be monitored by licensee corrective action programs. The area of configuration control will be included in the baseline inspection program as noted below and will assess human performance as well as other causes of configuration control deficiencies.

Licensed operator training program implementation in the area of mitigation of the potential for pressurized thermal shock (PWRs), water hammer within the RCS, and maintaining reactor coolant pump or recirculation pump seal cooling during off-normal conditions will be examined in a risk-informed manner through specific requirements in the baseline inspection program.

Severe Accident Management Guidelines (SAMGs) may include strategies for dealing with issues that could impact RCS integrity. They are considered under the emergency preparedness cornerstone.

C.4.3 Procedure Quality

Adequate procedures for routine operations, maintenance, surveillance testing, and emergency operations conditions are necessary to maintain assurance that LOCA frequency estimates remain relatively low.

The adequacy of routine operations and maintenance procedures that could affect the engineered strength margins of the RCS pressure boundary could appear as causal factors of deficiencies in other key attributes such as modification work quality, configuration control, and equipment and barrier performance. Thus, no specific measurement of routine procedure quality is warranted. This area will be monitored by licensee corrective action programs.

Changes to emergency operating procedures (EOPs) and to off-normal procedures invoked by EOPs in the area of mitigation of the potential for pressurized thermal shock (PWRs), water hammer within the RCS, and maintaining reactor coolant pump or recirculation pump seal cooling during off-normal conditions will be examined in a risk-informed manner through specific requirements in the baseline inspection program.

C.4.4 Configuration Control

Proper configuration control is necessary to maintain assurance that LOCA frequency estimates remain relatively low. Configuration control refers to maintaining operational control over physical conditions which, if such control is degraded, may result in a loss of RCS integrity. Inspection activities related to maintenance and operational realignments of the RCS during shutdown conditions will be performed in a risk-informed manner through specific requirements in the baseline inspection program.

Configuration control also includes maintaining operational control over RCS chemistry conditions (and possibly secondary chemistry conditions for PWRs) that could impact the engineered strength margin of RCS components. This area will be monitored by licensee corrective action programs.

C.4.5 Barrier and Equipment Performance

RCS leakage is the most direct measure of RCS barrier performance. All other key attributes under RCS integrity are aimed at measuring or inspecting areas that are known to contribute to the increased probability that RCS integrity could fail. An actual RCS leak is, by definition, a breach of RCS integrity and a direct indicator of the performance of the RCS pressure boundary. Research sponsored by the industry and NRC has determined that the RCS pressure boundary passive components have a high probability of experiencing a leak prior to a rupture (i.e., "leak-before-break" analysis). Therefore, two performance indicators have been identified that can offer an objective perspective on the probability of more catastrophic failure potential: the rate of occurrence and magnitude of small RCS pressure boundary leaks.

The condition of passive RCS pressure boundary components such as piping, welds, and valves is monitored by the licensee to maintain confidence in LOCA frequency estimates as degradation can potentially impact the RCS strength margins and the likelihood of an RCS pressure boundary rupture. A performance indicator has been proposed for this area (i.e., Inservice Inspection Results). In addition to this performance indicator or until the indicator is fully developed, the baseline inspection program will assess the effectiveness of the inservice inspection program in a risk-informed manner.

Active RCS pressure boundary components are defined here to include safety relief valves, power operated relief valves, and reactor coolant pump or recirculation pump seals and associated seal cooling equipment. Failure of active components can have a direct impact on RCS integrity. A high availability and reliability of the active components is expected through the licensee’s implementation of the maintenance rule. Any problems related to these components will be identified through NRC verification of the licensee’s implementation of the maintenance rule.

C.5 Key Attributes Affecting Containment

C.5.1 Design

The margins of safety in the containment design result in a containment ultimate pressure capacity substantially higher than design, and provide an inherent capability to withstand the extreme pressure loads associated with severe accident phenomena. The safety margins could be reduced if inadequate plant modifications are implemented. Therefore, it is important to assure that the containment structures and systems are maintained consistent with the original design. Design control issues stemming from deficient modifications will be identified by inspection of risk-significant plant modification packages and post-modification testing.

The structural integrity of the containment building and the operational capability of SSCs important to maintaining containment functionality were established through the original design and licensing review and confirmed through the pre-operational test and inspection program. This included conducting baseline integrated leak rate tests and system-level tests to confirm containment structural integrity, containment heat removal capabilities, and containment isolation capabilities. Periodic leak rate testing in accordance with Appendix J provides assurance that containment structures and components remain capable of resisting postulated design loads and preventing leakage in excess of technical specification limits (for design basis accident conditions). Continued operational capability will be reviewed through risk informed inspection of design features of containment systems not subject to periodic testing.

C.5.2 Human Performance

Human errors during routine operations and maintenance activities (e.g., errors affecting configuration control or equipment/barrier availability or reliability) can affect the functionality of the containment and potentially increase risk. The effectiveness of the control room operators and technical support center staff in maintaining containment integrity during response to an event will also impact risk. Issues related to human performance during routine operations and maintenance activities are expected to be identified and resolved by the licensee's corrective action program. Where significant problems in these areas are identified by the licensee corrective program or by other means, inspection followup of associated causal human performance deficiencies might be warranted and could be assessed in a reactive inspection. Issues related to performance under accident conditions will be identified through NRC observation of licensed operator training programs and through NRC's oversight of licensee emergency preparedness capabilities.

C.5.3 Procedure Quality

Inadequate procedures can complicate plant response by causing plant personnel to take inappropriate actions during plant operations, maintenance, testing, and emergency response. This can occur for reasons such as a missing step, ambiguous or confusing language or organization, or errors in the procedure stemming from inadequate supporting technical analyses.

The adequacy of routine operations and maintenance procedures that could affect containment functionality should be evident in activities under other key attributes such as modification work quality, configuration control, and equipment/barrier performance. Inspection will review the adequacy of test procedures to test those design functions being verified. No other specific measurement of routine procedure quality is needed. However, this area could be a root cause of inadequate performance in configuration control or equipment/barrier availability/reliability. Where significant problems in these areas are identified by the licensee corrective program or by other means, inspection of associated causal procedure quality deficiencies might be warranted and could be assessed in a reactive inspection.

The quality of EOPs and other off-normal procedures invoked by the EOPs is central to assuring that appropriate actions will be taken by the operator to protect and preserve containment integrity under accident conditions. Procedures which could significantly impact containment functionality and offsite risk include those related to depressurizing the RCS; controlling containment pressure, temperature, and hydrogen concentrations using engineered safety features; flooding containment; and venting containment. Problems related to procedure quality will be identified through risk-informed inspection of licensee EOP modification packages.

C.5.4 Configuration Control

Inadequate control of the lineup of containment penetrations and containment-related SSCs could decrease or directly compromise containment functionality. Examples of configuration control problems include mispositioning containment isolation valves, leaving containment penetrations open or unable to be rapidly closed during shutdown when needed, or inadvertently isolating containment heat removal systems. Performance indicators would not be expected to be useful for trending significant configuration control problems because such problems occur rarely. Problems related to maintaining the risk-significant containment SSCs in their proper condition will be identified by the licensee's corrective action program, and by inspection of containment configuration during risk-significant evolutions.

It is also important that the plant be operated within containment design limits, such that the containment is in a condition ready to accommodate a design basis accident or severe accident. Significant deviations from design limits are not expected since the plant is equipped with various design features (e.g., alarms and interlocks) to protect key systems/functions and is operated in accordance with technical specifications. Also, the design of the containment structure contains substantial margins such that modest deviations from design limits will not impact containment functionality. However, extreme deviations of certain containment parameters (such as low suppression pool level and loss of an inerted environment) could threaten containment integrity. Inspection is not required because compliance with technical specification requirements for containment parameters is adequate. Noncompliance would generally be indicated by control room indications and alarms and would require reporting and prompt action to address.

C.5.5 Barrier and Equipment Performance

Containment integrity can be inferred if all of the following conditions are met for the risk-significant penetrations: (1) all normally closed containment isolation valves and hatches are in the appropriate position, (2) isolation valves and penetrations which are permitted to be open during power or shutdown can be closed in a timely manner, and (3) the total leak rate is within acceptable limits. Failure to close containment penetrations or excessive leakage through large containment penetrations could result in a loss of containment functionality and a risk-significant release to the environment. A high availability and reliability of the containment isolation function (and associated containment isolation valves and penetrations) is expected through implementation of the licensee's maintenance program. Any problems related to containment isolation should be identified through NRC verification of the licensee's implementation of the maintenance rule. Finally, the leak rate for containment will be trended by a performance indicator.

Given that containment isolation is achieved, certain SSCs are required to assure that containment functional integrity will be maintained during design basis and severe accidents (e.g., containment sprays and hydrogen control). Failure of these SSCs could lead to containment over-pressure or other containment release modes. A high availability and reliability of the containment-related SSCs is expected through the licensee's implementation of the maintenance rule. Any problems related to containment-related SSCs will be identified through NRC verification of the licensee's implementation of the maintenance rule.

C.6 Performance Indicators - Barrier Integrity

Reactor Coolant System (RCS) Activity Level

This performance indicator provides an objective means of measuring fuel cladding integrity in the equipment/barrier performance key attribute area. It also provides a measure of performance of certain aspects of the other key attribute areas. An increase in RCS radioactivity level can be directly correlated to the performance (integrity) of the fuel cladding barrier since perforation of the cladding will result in the release of fission products to the RCS. Monitoring RCS activity is important from a risk-informed perspective since a failure of fuel cladding is by definition a breach of one of the three barriers to fission product release in the "defense-in-depth" protection scheme. This performance indicator is the maximum calculated reactor coolant system specific activity per month. The data required to develop this performance indicator is already being generated frequently at each reactor facility through analysis of RCS samples as required by technical specifications. The thresholds for this indicator have a regulatory basis which is only indirectly linked to a risk basis. They will be set at 50 percent and 100 percent of the technical specification limit based an expert panel process using the NEI proposal as an input. Additional NRC attention is warranted to determine the cause of increased RCS activity at a level of 50% of the technical specification limit (Increased Regulatory Response Band). Individual plant technical specifications would require plant shutdown within a short time after RCS activity exceeds the technical specification limit (Required Regulatory Response Band).

One limitation of this performance indicator is that it will only indicate when fuel cladding has actually failed, and will not indicate a slow degradation in cladding condition prior to penetration. In spite of this limitation, this type of monitoring is sufficient to indicate the overall "health" of the nuclear fuel cladding. If unacceptably high radioactivity levels are indicated in the RCS, individual licensee technical specifications would require that appropriate remedial actions be implemented before an unacceptable degree of fuel cladding degradation occurred.

Verification activities associated with this performance indicator could be conducted by performing periodic observations of primary water chemistry sampling and analysis to ensure that licensee personnel are accurately collecting and recording the necessary data.

RCS Leakage

Two performance indicators are proposed to be used to measure equipment and barrier performance for the RCS. The first direct measure is "RCS leak rate". This performance indicator is the maximum calculated reactor coolant system leak rate per month. The data required to develop this performance indicator is already being generated frequently at each reactor facility through RCS leakage determination as required by technical specifications. This indicator relies upon existing technical specification definitions (identified leakage plus unidentified leakage) and therefore needs no new definition of terms or verification strategy. The thresholds for this indicator have a regulatory basis as opposed to a direct risk basis. They will be set at 50 percent and 100 percent of the technical specification limit based an expert panel process using the NEI proposal as an input. Additional NRC attention is warranted to determine the cause of elevated RCS leakage at a level of 50% of the technical specification limit (Increased Regulatory Response Band). Individual plant technical specifications would require plant shutdown within a short time after the RCS leak rate exceeds the technical specification limit (Required Regulatory Response Band).

The second direct measure of RCS barrier integrity could be defined as, "Occurrence rate of individual RCS pressure boundary (as defined by technical specifications) leaks, measured on a per fuel cycle basis, that contribute to identified RCS leakage, that are not primary-to-secondary leakage, and that exist when RCS integrity is required by technical specifications." This performance indicator requires further development.

RCS Inservice Inspection Results
A potential performance indicator to monitor the degree of degradation of the RCS barrier could be "the percentage of individual inservice inspection tests performed within [TBD] that require disposition against ASME acceptance standards" (steam generator tube inspections are treated separately under the initiating events cornerstone). Such an indicator can be objectively derived and a threshold set that is related to historically good industry performance. By using a percentage indicator, instead of an absolute number indicator, it is less likely to influence the assessment of non-destructive examination (NDE) examiners as the number count of flaw indications increases. Verification and validation of this performance indicator should include ensuring that industry operating experience is being applied to the selection of areas for NDE. This performance indicator requires further development.

Containment Leakage
The estimated "as-found" integrated leak rate for the containment provides a reasonable indication of what actually existed during operation, and provides an indication of the leak-tight integrity of the containment barrier. Measurement data would be based on the last integrated leak rate test result, modified by the results of subsequent local leak rate tests. The data would be reported as a fraction of the design basis leak rate (L_a). Licensees currently collect this data as required by 10 CFR Part 50 Appendix J. Data would be reported quarterly although, in some quarters, no new data would be collected at a particular site. The threshold for increased regulatory oversight (Increased Regulatory Response Band) would have a regulatory basis and would be set at a leak rate corresponding to the plant’s technical specification limit for allowable containment leakage. Use of the technical specification value provides considerable margin since offsite risk is not significantly increased until the containment leak rate approaches 100 percent per day (i.e., several orders of magnitude greater than L_a. Leakage at the technical specification limit is not risk significant, so this threshold provides an element of defense in depth. A threshold for the Required Regulatory Response Band is not proposed since licensees are expected to make repairs to the containment and to reduce the leak rate below L_a in a short time or shut down in accordance with technical specification requirements.

Two limitations with this performance indicator should be noted:
(1) "As-found" leak rate data is not collected in a consistent manner at all plants. Specifically, some plants perform the Type C tests at the end rather than at the beginning of the refueling outage. The leak rate data for those plants may not reflect the actual leak rate that existed during power operation, particularly if the isolation valves are cycled during the outage. Some changes to licensee practices may be needed to achieve consistency.
(2) The data obtained from integrated and local leak rate tests is gathered relatively infrequently. In accordance with Appendix J, licensees are required to perform integrated leak tests (Type A tests) on a frequency of 3 tests every 10 years, and to leak test Type B and Type C components during each reactor shutdown for refueling, but in no case at intervals greater than 2 years. Licensees adopting Option B of Appendix J can extend the integrated leak test frequency to one test every 10 years, and extend the test interval up to 60 months for Type B penetrations (except personnel airlocks) and Type C components (except main steam and feedwater isolation valves in BWRs, and containment purge and vent valves in PWRs and BWRs). The extended test interval for those excepted components would be limited to 30 months. Thus, depending on the licensee’s test program, updates to the performance indicator would occur on an infrequent basis.

C.7 Inspection Areas - Barrier Integrity

C.7.1 Inspection Areas - Fuel Cladding Integrity

Configuration Control
In order to provide confidence in the defense in depth element provided by the fuel cladding barrier, certain inspections are needed to supplement the performance indicator. Fuel cladding degradation can result from both inadequate human and equipment performance. Control rod configurations (patterns) must be properly established and maintained to ensure that abnormal alignments do not result in challenges to core thermal limits and ultimately fuel cladding integrity. Reactivity manipulations must be conducted in a well-controlled and deliberate manner to provide assurance that reactor power operation will remain within the limits established by technical specifications. Reactivity control systems must also be properly configured to prevent and/or mitigate adverse reactivity transients and neutron flux distributions. Performance-based inspection activities to address these issues include:

1. Periodic observations of licensed operators during the conduct of reactivity manipulations (e.g. to ensure adherence to vendor-provided fuel preconditioning limits). Inspection in this area should be conducted during significant reactivity manipulations (e.g. >20% in the power range), and during plant startups and shutdowns.

2. Evaluations of maintenance activities associated with reactivity control systems (e.g. control rod drives, rod block monitors, rod worth minimizers, etc.) to ensure that they remain capable of performing their functions following the work. Periodic observation of instrument channel calibrations and functional tests of reactivity control equipment should also be included. Control rod drive mechanism work, including hydraulic control units for BWRs, should also be periodically assessed.

3. Verifications of nuclear instrument performance to ensure that they are properly calibrated and provide protection signals at the proper set points.

4. Reviews of computer-generated thermal limit reports to verify that defined safety limits and operating margins are preserved.

Corrective Action Program
In addition, deficiencies associated with certain other activities which could affect fuel cladding integrity and reduce confidence in the measure of defense in depth which it provides should be monitored during the planned baseline inspection of licensee corrective action programs. Possible focus areas include errors associated with:
- Core design analysis
- Start up physics testing
- Human performance (e.g. procedure adherence, etc.)
- Procedure quality
- Primary water chemistry control
- Refueling
- Loose parts monitoring and foreign material exclusion

C.7.2 Inspection Areas - Reactor Coolant System

Design Control
Maintaining continued confidence in LOCA frequency estimates will include measuring the quality of design modification or temporary modification activities that could increase the probability of an RCS pressure boundary rupture. The definition of RCS pressure boundary used here, for inspection purposes, extends beyond the passive pressure retaining piping, valves, and other components covered by ASME

code requirements. It also includes active components such as reactor coolant pump or recirculation pump seals and safety relief valves.

As a measure of the quality of design control as it is related to the RCS barrier, an inspection should review a sample of proposed risk-significant modification packages that affect the RCS pressure boundary, including active components. These will include those which could simultaneously impact both RCS integrity and mitigation system reliability or performance. As opportunities occur to observe the quality of work in progress in this area, including post-modification testing, inspectors should assess the ability of the licensee to maintain the design pressure retention capability of the RCS pressure boundary, which forms the basis for assurance that LOCA frequency estimates remain low. Because of their potential importance to risk during station blackout (SBO) conditions, plants having relatively significant contributions to core damage frequency from SBO, reactor coolant pump or recirculation pump seal replacement or modification should receive high priority, particularly for those seals whose design has not been enhanced for high temperature service (e.g., Westinghouse high temperature RCP seals). In addition, because the presence of pressure relief valves (e.g., code safety valves and power operated relief valves) increases the opportunity for LOCAs due to failures to reseat following lifting, replacement or modification of these components should also receive high priority. The inspection procedure for this area should provide historical insights of causes for pressure boundary failures so as to alert the inspector to the most likely problem areas. For example, for passive components attention should be paid to modifications that might increase mechanical fatigue (e.g., small diameter piping attached to much larger diameter piping), or thermal fatigue (e.g., stratification of liquids or turbulent mixing of hot and cold fluids), or use of material compositions that could increase corrosion susceptibility (e.g., IGSCC, PWSCC), or that might increase the probability of water hammer*. Similar historical insights should be collected for pump seals and relief valves and used as inspection guidance.

* Welding Research Council Bulletin #382, June 1993, "Nuclear Piping Criteria for Advanced Light-Water Reactors, Volume 1 - Failure Mechanisms and Corrective Actions, ISSN 0043-2326, provides an excellent overview of historical insights for piping degradation mechanisms.

Human Performance

As a measure of post accident or event human performance, the inspection program should include licensed operator requalification program implementation with emphasis on simulator observation in the areas of mitigation of potential for pressurized thermal shock (PWRs), water hammer within the RCS, and maintaining reactor coolant pump or recirculation pump seal cooling during off-normal conditions.

Emergency Operating Procedures

LOCAs can occur as a consequence of certain other accident sequences. The contribution to core damage frequency from these consequential LOCAs can vary dramatically between plants. Usually an implicit assumption is that emergency operating procedures (EOPs) are relatively effective in preventing serious degradation of the RCS pressure boundary during such sequences. These operator actions include those that mitigate the impact to passive components (e.g., piping) from pressurized thermal shock and mechanical shock due to water hammer, and to active components such as operator actions to restore cooling to reactor coolant or recirculation pump seals following a station blackout.

To measure the quality of emergency operating procedures as they relate to the RCS barrier, inspection should sample modification packages for emergency operating procedures (and off-normal procedures which are referenced) that could affect the RCS pressure boundary, including active components. Although this review should focus on the modification, it should also include a broad review of the underlying EOP strategy in the area affected by the modification to ensure that the strategy remains sound and in accordance with its intended objectives as described in licensee EOP basis documents.

Configuration Control

Configuration control refers to maintaining system alignment control over active components of the RCS pressure boundary (e.g., isolation valves, PORVs, pump seals) which, if such control is degraded, may result in a loss of RCS integrity. This is not generally modeled in risk assessments of at-power conditions. However, inter-system LOCAs (ISLOCAs) are often modeled as catastrophic failures of normally closed valves whose function is to prevent high pressure RCS coolant from over-pressurizing low pressure components such as those associated with decay heat removal systems. Although such events have a very low estimated occurrence frequency, the resulting coolant loss is not recoverable in the containment and therefore not available for long term core and containment heat removal. This makes ISLOCA contribution to risk very sensitive to the valve failure frequency estimate. However, spontaneous catastrophic failure of a valve is not nearly as likely as an operator mis-positioning event. Such operator-induced events would be more likely during shutdown plant conditions when maintenance and system re-alignments are in progress. Therefore, the risk significance of ISLOCA events is increased during periods of operator manipulation of active pressure boundary components and in particular where an ISLOCA could degrade mitigation equipment capability.

The baseline inspection program should assess configuration control as it relates to RCS barrier integrity during shutdown operations. This should include RCS and associated/attached systems (e.g., Low Temperature Overpressure Relief Valves) configuration and manipulations to assure that RCS integrity is maintained and controlled.

Barrier and Equipment Performance

The rate at which RCS pressure boundary leaks (ASME definition) occur is a proposed performance indicator, which in combination with an RCS leak rate performance indicator gives a complete picture of the RCS barrier performance. However, until the "rate of leaks" indicator is fully developed, inspection is warranted to monitor the rate and cause (if known) of such leaks and to assess the adequacy of licensee corrective actions.

Similarly, until the inservice inspection performance indicator is fully developed, inspection is warranted to assess the adequacy of the inservice inspection program scope, including the use of plant-specific risk insights and industry operating experience

As another aspect of equipment performance, reactor coolant pump or recirculation pump seals and associated cooling equipment and RCS pressure relief valves should be inspected through verification inspections of the licensee’s implementation of the maintenance rule. The focus of these inspections should be on performance that may indicate an increasing probability of RCS pressure boundary failure (e.g., pump seal failure, stuck open relief valve).

Corrective Action Program

In addition, deficiencies associated with certain other activities which could affect the RCS integrity and reduce confidence in the measure of defense in depth which it provides should be monitored during the planned baseline inspection of licensee corrective action programs. Possible focus areas include errors associated with:

- Human performance deficiencies related to routine maintenance and surveillance testing of the RCS
- Adequacy of routine operations and maintenance procedures that could affect the engineered strength margins of the RCS pressure boundary

- RCS chemistry conditions

C.7.3 Inspection Areas - Containment

Design Control
As a measure of how design control affects the containment barrier and in order to ensure that the design basis and PRA assumptions remain valid, inspectors should perform a design review of a sample of risk-significant modifications or temporary modifications. In addition, for this limited set of modifications, inspectors should conduct a performance-based inspection of the post-modification testing.
Priority should be given to review of modifications that may:

- adversely impact the functionality of systems important to long term containment pressure control and degraded core hydrogen control (e.g., sprays, Mark I hardened vent, isolation condenser, igniters)
- increase the likelihood or magnitude of steam/fission products bypassing the suppression pool or ice condenser (e.g., vacuum breakers, ice condenser components)
- reduce the availability/reliability of isolating large diameter containment penetrations (> 2 inches) which connect to the containment airspace (e.g., purge/vent valves, vacuum breakers, actuation system)
- extend the time required to achieve containment closure during shutdown
- reduce the containment ultimate pressure capacity or introduce new containment failure modes (temporary containment equipment hatches)

The inspector should consult the plant-specific risk study to identify the most risk-significant containment-related SSCs for a particular plant, and to establish a basis for selecting the design changes to be reviewed. Continued operational capability will be reviewed through risk informed inspection of design features of containment systems not subject to periodic testing.

Human Performance
As a measure of how human performance in an accident or event situation affects the containment barrier, the NRC should continue to conduct inspections of licensed operator training to confirm that risk-significant human actions are addressed within the training program, and that control room crews are able to effectively carry out the risk-significant human actions during simulated accidents involving these actions. The NRC should also confirm the adequacy of the licensee's self-assessment of its severe accident management (SAM) capabilities as part of NRC's oversight of licensee emergency preparedness programs.

Test Procedures
Inspection is needed to confirm that test procedures adequately test those system design features being verified. Those design features not verified by routine testing will be subject to risk informed inspection.

Emergency Operating Procedures
Inspection is needed to confirm the quality of EOPs which affect the containment boundary. The quality of the plant-specific EOPs was verified through the NRC's EOP inspection program conducted in 1988-1991. Using the current EOPs as a baseline, information is needed only on risk-significant changes to the procedures. The inspector should sample EOP modification packages that could affect containment integrity, isolation capabilities or SSCs important to LERF (such as ATWS response, containment venting, and manual depressurization). Although this review should focus on the modification, it should also include a broad review of the underlying EOP strategy in the area affected by the modification to ensure that the strategy remains sound and in accordance with its intended objectives as described in licensee EOP basis documents.

Configuration Control
Inspection is recommended to confirm the adequacy of configuration control as it affects the containment boundary and SSCs important to LERF. The risk-significant penetrations would be identified based on the plant-specific risk study, and are expected to comprise a small fraction of the total containment penetrations. The inspector should verify proper containment configuration during risk-significant evolutions (e.g. PWR mid-loop operation, BWR cavity drain downs, etc.). This should include a review of the licensee's provisions for achieving containment closure in a timely manner (i.e., prior to RCS steaming) during periods when the containment is permitted to be open. Inspections in this area are important because the high safety significance of these activities.

Barrier and Equipment Performance
Inspection is needed as a measure of equipment performance related to the containment barrier. Reliability and availability data for containment penetrations which constitute the major pathways for release to the environment provides an indicator of the reliability of the containment isolation function. As part of the baseline inspection program for maintenance rule oversight, the inspector should perform a periodic review of the availability and reliability information for those penetrations important to LERF. These penetrations would be identified based on the plant-specific risk study, and are expected to comprise a small fraction of the total containment penetrations. The penetrations are expected to include the large diameter piping penetrations through which the containment air space or reactor coolant system could communicate with the outside environment (e.g., purge/vent penetrations and MSIVs), personnel airlocks, and equipment hatches.

The inspector should also review the information from the licensee's maintenance program for each SSC judged to be important for controlling the LERF. The risk-significant SSCs are containment- and plant-specific and should be selected on the basis of their importance to large release frequency in the plant-specific risk study. The SSCs which should be considered for monitoring include those critical for:
- short and long term pressure control (e.g., containment spray and fan coolers in PWRs; suppression pool cooling, isolation condenser, drywell/wetwell sprays, and drywell/wetwell vents in BWRs), and
- degraded core hydrogen control (i.e., hydrogen igniters for ice condenser and Mark III containments and inerting in Mark I and II containments).

The necessary reliability and availability data for the major containment isolation components and SSCs important to LERF are expected to be available from the licensee's implementation of the maintenance rule. During the maintenance rule baseline inspection, the inspector should verify that the licensee accurately collects and assesses the needed data.

Corrective Action Program
In addition, deficiencies associated with certain other activities which could affect containment functionality and reduce confidence in the measure of defense in depth which it provides should be monitored during the planned baseline inspection of licensee corrective action and self-assessment programs. Possible focus areas include licensee follow-up of:
- Instances in which measured leakage is found to exceed L_a
- Human errors impacting containment integrity that are identified as root causes of problems in other areas
- Procedure deficiencies impacting containment performance that are identified as root causes of problems in other areas
- Failures to maintain the proper status of risk-significant containment isolation valves and penetrations
- Failures to maintain containment parameters within design limits

TABLE C1 - SUMMARY OF PROPOSED INDICATORS FOR THE BARRIER INTEGRITY CORNERSTONE

Measure	Purpose	Indicator	Thresholds
RCS Activity	To provide indication of fuel barrier integrity and occurrence of cladding failure	Maximum calculated activity level (microCuries per gram dose equivalent Iodine-131) per month	50% of TS Limit; 100% of TS Limit
RCS Leak Rate	To provide indication of the potential for a breach of the RCS	Maximum calculated leakage rate (gallons per minute)per month (identified plus unidentified)	50% of TS Limit; 100% of TS Limit
RCS Leak Occurrence Rate **(Future)**	To provide a measure of the frequency of RCS leaks	Occurrence rate of individual RCS pressure boundary leaks (as defined by technical specifications), measured on a per fuel cycle basis, that contribute to identified RCS leakage, that are not primary-to-secondary leakage, and that exist when RCS integrity is required by technical specifications	TBD
RCS Inservice Inspection Results **(Future)**	To provide indication of the potential for RCS failure	The percentage of individual inservice inspection tests performed within [TBD] that require disposition against ASME acceptance standards	TBD
Containment Leakage	To provide indication that containment leakage will remain below levels corresponding to a large radiological release, given that containment closure is achieved	Total leakage (fraction of the design basis leak rate, L_a) from containment as determined from the last integrated leak rate test, updated by the "as found" results of subsequent local leak rate tests required by 10 CFR 50, Appendix J	100% of TS Limit with respect to L_a

TABLE C2 - BARRIER INTEGRITY KEY ATTRIBUTES AND MEANS TO MEASURE

Barrier - Key Attributes	Areas to Measure	Means to Measure	Discussion
Fuel Cladding Integrity - Design Control	Core Design Analysis	Performance Indicator (RCS Activity)	Design errors could lead to cladding defects or failures, the effect of which would be seen in the performance indicator. Gap release is assumed in certain design basis accidents. Design errors would not be expected to cause a risk-significant increase in the gap release. Errors in the core design analysis should be detected during start up physics testing and data review
	Physics Testing	Corrective Action Program	Physics testing is conducted while low in power; where design errors are not likely to challenge cladding integrity. Should significant problems be identified, shutdown and corrective actions would be accomplished.
Fuel Cladding Integrity - Human Performance	Procedure Adherence	Performance Indicator (RCS Activity), Corrective Action Program	Failure to adhere to procedures, assuming that it results in adverse consequences, would be seen in the RCS activity performance indicator or should be identified as a root cause of problems measured in other key attribute areas (see Fuel Cladding Integrity - Configuration Control)
	Foreign Materials Exclusion (FME)	Performance Indicator (RCS Activity), Corrective Action Program	The corrective action program would be expected to identify and correct FME problems. In some cases, FME problems could lead to cladding defects which would be identified by the RCS activity performance indicator.
Fuel Cladding Integrity - Procedure Quality	Quality of Procedures Which Could Impact Cladding	Performance Indicator (RCS Activity), Corrective Action Program	In the worst case, inadequate procedures could result in fuel cladding damage, which would be reflected in the RCS activity performance indicator. Less significant procedure deficiencies should be captured as root causes of problems measured in other key attribute areas

Barrier - Key Attributes	Areas to Measure	Means to Measure	Discussion
Fuel Cladding Integrity - Configuration Control	Reactivity Control	Inspection	Monitor those activities which could lead to fuel cladding degradation. Abnormal control rod alignments or reactivity manipulations during plant operation can result in reduction in margins to core thermal limits and even challenge thermal limits during transients, leading to cladding degradation or failure. Misconfigured or malfunctioning reactivity control systems may fail to prevent or mitigate areas of unacceptably high neutron flux in the core which could lead to fuel cladding damage. Cladding perforation is by definition a breach of the fuel barrier and a reduction in the defense-in-depth for prevention of fission product release to the environment.
	Primary Chemistry Control	Corrective Action Program, Performance Indicator (RCS Activity)	Problems resulting from inadequate water chemistry controls tend to develop slowly and should be adequately identified and resolved by effective implementation of licensee self-assessment and corrective action programs. The RCS activity performance indicator would provide a back-up.
	Core Loading	Corrective Action Program, Performance Indicator (RCS Activity)	Fuel loading errors committed during the refueling process should be detected while very low in power during start up physics testing. Improperly placed or oriented fuel assemblies can lead to localized areas of high neutron flux with adverse consequences. Fuel assembly mispositioning errors should be identified during independent verification of the core configuration prior to vessel head re-installation. The licensee’s corrective action program is expected to identify and resolve this type of problem, as well as problems involving cladding damage during handling. The RCS activity performance indicator would provide a back-up.
Fuel Cladding Integrity - Equipment/ Barrier Performance	Reactor Coolant System (RCS) Activity	Performance Indicator (RCS Activity)	RCS radioactivity level measurements provide a reliable means of indicating when nuclear fuel cladding has been compromised, resulting in a direct and objective measure of the integrity of the fuel cladding barrier. This PI is important from a risk-informed perspective since a failure of fuel cladding is by definition a breach of one of the three barriers to fission product release in the "defense-in-depth" fission product release protection scheme.

Barrier - Key Attributes	Areas to Measure	Means to Measure	Discussion
	Loose Parts	Performance Indicator (RCS Activity),	Besides FME issues (described in the "Fuel Cladding Integrity - Human Performance" key attribute above), loose parts can be introduced into the reactor vessel by poor maintenance practices or failures of internal structural components. In some cases, loose parts could lead to cladding defects which would be identified by the RCS activity performance indicator.
Reactor Coolant System Integrity - Design Control	Modifications	Inspection	Review proposed permanent or temporary modification packages for risk-significant SSC's, including the associated 10 CFR 50.59 safety evaluations. This effort should ensure that design bases and risk analyses assumptions are preserved. Inspection should also focus on post-modification testing to verify that "as-left" equipment or barrier performance is satisfactory. The scope of this effort should focus on the most risk-significant modifications, for example those which could simultaneously impact both RCS integrity as well as mitigation system performance or reliability.
Reactor Coolant System Integrity - Human Performance	Routine Performance	Corrective Action Program	Errors (including failures to adhere to established procedures) should be captured as root causes of problems measured in other key attributes
	Post-Accident or Event Performance	Initial Operator Exams and Requalification Program Inspections	Observe licensed operator initial and requalification examinations with focus on actions which are designed to protect the integrity of the RCS barrier. These actions include those that mitigate the impact to passive components (e.g. piping) from direct thermal impacts (e.g. pressurized thermal shock) and mechanical shocks (e.g. water hammer), and actions to restore cooling to reactor coolant/recirculation pump seals during conditions affecting the adequacy of cooling to these seals.
Reactor Coolant System Integrity - Procedure Quality	Routine procedures	Corrective Action Program	Operations, maintenance and surveillance procedure deficiencies should be captured as root causes of problems measured in other key attributes

Barrier - Key Attributes	Areas to Measure	Means to Measure	Discussion
	Emergency Operating Procedures (EOP) and Related Off-Normal Procedures	Inspection	Focused review of proposed risk-significant changes to EOPs. The quality of EOPs and other off-normal procedures go hand-in-hand with effective human performance to provide adequate assurance that RCS pressure boundary components will be protected during accidents or events involving these procedures. The quality of these procedures is equally risk significant as that noted for the human performance area discussed above. During the review of proposed EOP changes, consider the conduct of a broader review of the subject EOP to ensure that the overall accident mitigation strategy is still valid.
Reactor Coolant System Integrity - Configuration Control	System Alignment	Inspection	Periodically verify during plant shutdown periods (when operator manipulation of RCS pressure boundary components like isolation valves is most frequent) that the configuration of the RCS and connected systems is properly maintained. The consequences of mis-positioned RCS boundary valves resulting in a LOCA can be high when the resulting coolant loss is not recoverable in the containment and therefore not available for long term core and containment heat removal.
	Primary and Secondary Chemistry Control	Corrective Action Program	Problems resulting from inadequate water chemistry controls tend to develop slowly and should be identified and resolved by internal licensee processes
Reactor Coolant System Integrity - Equipment/ Barrier Performance	Reactor Coolant System Leakage	Performance Indicator (RCS Leak Rate)	Monitor the extent of RCS leakage. An actual RCS leak is, by definition, a breach of RCS integrity and a reduction in the defense-in-depth for protection against fission product release. RCS leakage is a direct indicator of the performance of the RCS pressure boundary. Research has determined that the RCS pressure boundary has a high probability of experiencing a leak prior to a rupture (i.e. "leak-before-break"). Therefore, the extent of such leaks offers an objective perspective on the probability of a more catastrophic failure.

Barrier - Key Attributes	Areas to Measure	Means to Measure	Discussion
		Performance Indicator (RCS Leak Occurrence Rate) (Potential)	Monitor the rate of occurrence of RCS pressure boundary leaks. RCS pressure boundary leaks, by definition, are breaches of RCS integrity and reduce defense-in-depth for protection against fission product release. Research has determined that the RCS pressure boundary has a high probability of experiencing a leak prior to a rupture (i.e. "leak-before-break"). Therefore, the rate of occurrence of such leaks offers an objective perspective on the probability of a more catastrophic failure.
		Inspection	Until the above potential performance indicator is available for rate of occurrence of RCS pressure boundary leaks, the baseline inspection program will monitor the rate and cause (if known) of such leaks and assess the adequacy of licensee corrective actions.
	Inservice Inspection (ISI) Results	Inspection	ISI programs, when effectively implemented, provide a proactive means to assess the overall integrity of the RCS. Emphasis will be placed on the use of industry operating experience to assess the adequacy of the inservice inspection program scope, including the use of plant-specific risk insights.
		Performance Indicator (RCS ISI Results) (Potential)	Monitor the number of RCS defects identified during licensee ISI. Implicit in the generally low LOCA frequency estimates resulting from plant risk assessment studies is the expectation that effective quality assurance activities (such as ISI) will monitor and maintain the engineered strength margins of the reactor coolant pressure boundary. A relatively large number of identified defects resulting from ISI would indicate either a robust ISI program, deficient RCS design or construction, or poor RCS pressure boundary maintenance.
	Active RCS Component Performance	Maintenance Rule Verification	Inspection should provide oversight of the licensee’s implementation of the maintenance rule, which includes monitoring the performance of reactor coolant or recirculation pump seals, safety/relief valves, etc. Poor performance associated with the active RCS components could invalidate the assumptions made in risk assessment studies and increase the potential for LOCAs.

Barrier - Key Attributes	Areas to Measure	Means to Measure	Discussion
Containment Integrity - Design Control	Structural Integrity	Performance Indicator (Containment Leakage)	Established during the initial licensing and pre-operational testing and inspection process; continuing adequacy in this area is assessed through inspection of related modifications (see below) and is monitored by leak rate testing (the performance indicator is described below).
	Operational Capability	Inspection	Established during the initial licensing and pre-operational testing and inspection process. Continuing adequacy in this area is assessed through inspection of related modifications (see below) and inspection of design features of containment systems not subject to periodic testing (see test procedures below).
	Modifications	Inspection	Review proposed permanent or temporary modification packages for risk-significant SSC's, including associated 10 CFR 50.59 safety evaluations, to ensure that design bases and risk analyses assumptions are preserved. Inspection should also focus on post-modification testing to verify that "as-left" equipment or barrier performance is satisfactory.
Containment Integrity - Human Performance	Routine Performance	Corrective Action Program	Human performance errors during routine operations, maintenance, and surveillance (including failures to adhere to established procedures) should be captured as root causes of problems measured in other key attributes
	Post-Accident or Event Performance	Initial Operator Exams and Requalification Program Inspections	Continue to assess licensed operator training, with focus on actions design to protect containment integrity. Risk studies indicate that certain operator actions can have a significant impact on plant risk. In BWRs these include actions to inhibit the automatic depressurization system and subsequently depressurize the RCS manually, align suppression pool cooling, control reactor level during an ATWS, and vent the containment. For PWRs these include actions to switch over from the injection to the recirculation phase of core cooling, feed and bleed using HPI and PORVs, and recover normal and emergency power.

Barrier - Key Attributes	Areas to Measure	Means to Measure	Discussion
Containment Integrity - Procedure Quality	Routine Operations, Maintenance and Surveillance Procedures	Inspection	Procedure deficiencies should be captured as root causes of problems measured in other key attributes. Inspection will confirm whether test procedures adequately test those system design features being verified. Important design functions not verified by testing will be subject to risk informed inspection (see design - operational capability, above)
	Emergency Operating Procedures	Inspection	Focused review of proposed risk-significant changes to emergency operating procedures. During this review of proposed EOP changes, consider the conduct of a broader review of the subject EOP to ensure that the overall accident mitigation strategy is still valid.
Containment Integrity - Configuration Control	Lineup of Containment Penetrations and SSCs Important to LERF	Corrective Action Program	Errors in maintaining the proper status of risk-significant containment penetrations and SSCs should be infrequent and identified via control room alarms and indications and routine surveillances
		Inspection	Verify that the containment is in the proper configuration and that open penetrations can be closed in a timely manner during risk-significant evolutions (e.g. "mid-loop" operation with fuel in the vessel at a PWR). Since defense-in-depth protection against a fission product release is already reduced in these circumstances, added assurance of the viability of timely and effective containment isolation is needed
	Containment Design Parameters Maintained	Corrective Action Program	Errors in maintaining the proper containment design parameters, many established by technical specifications (e.g. torus level in BWR), should be infrequent and easily identified (i.e. via control room alarms and indications).
Containment Integrity - Equipment/ Barrier Performance	Steam Generator Tube Integrity and ISLOCA Prevention	Covered under equipment performance attribute of the Initiating Events Cornerstone and configuration control attribute of RCS barrier	

Barrier - Key Attributes	Areas to Measure	Means to Measure	Discussion
	Containment Isolation Systems Reliability and Availability	Performance Indicator (Containment Leakage)	Monitor the "as-found" containment leak rate data. "As-found" data is important because it provides an objective and reasonable indication of what actually existed during previous plant operation. The PI would be reported as the combined total leak rate of all the penetrations, as a fraction of the site-specific L_a.
		Maintenance Rule Verification	Inspection should provide oversight of the licensee's implementation of the maintenance rule, which includes monitoring the performance of containment isolation SSCs which constitute major release pathways to the environment (i.e. important to LERF).
	Risk-Important Support Systems Availability and Reliability	Maintenance Rule Verification	Inspection should provide oversight of the licensee's implementation of the maintenance rule, which includes monitoring the performance of containment support systems which could adversely impact the functionality of the containment. For example, these systems could include containment spray and hydrogen ignitors.

Appendix D
Emergency Preparedness Cornerstone

GENERAL DESCRIPTION

Emergency Preparedness (EP) is the final barrier in the *defense in depth* approach to safety that NRC regulations provide for ensuring the adequate protection of the public health and safety. Emergency Preparedness is a fundamental cornerstone of the Reactor Safety Strategic Performance Area. 10 CFR Part 50.47 and Appendix E to Part 50, define the requirements of an EP program and the licensee commits to implementation of these requirements through an Emergency Plan (the Plan).

Statement of Objective

Ensure that the licensee is capable of implementing adequate measures to protect the public health and safety in the event of a radiological emergency.

Desired Result / Performance Expectation

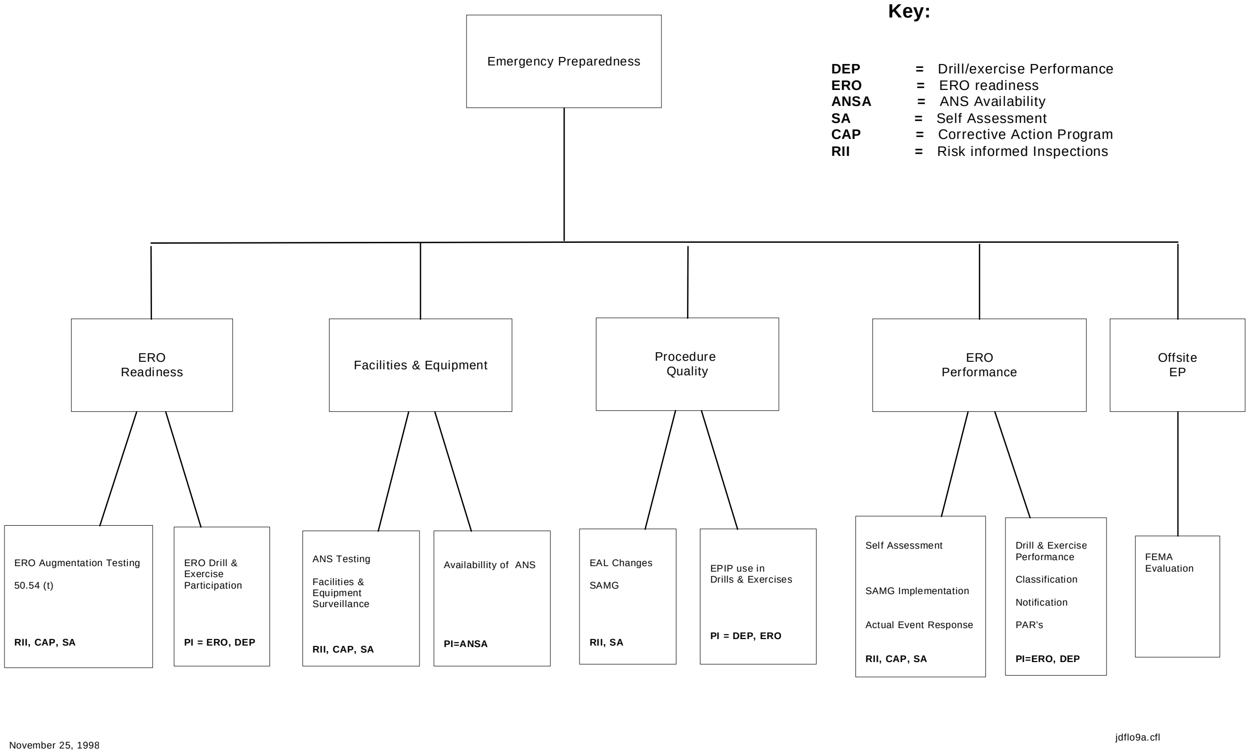
Demonstration that reasonable assurance exists that the licensee can effectively implement its emergency plan to adequately protect the public health and safety in the event of a radiological emergency.

KEY ATTRIBUTES OF LICENSEE PERFORMANCE THAT CONTRIBUTE TO EMERGENCY PREPAREDNESS

Measures taken to protect the public from the effects of a radiological emergency must necessarily involve action by both licensee and State and local governmental authorities in the vicinity of the reactor. The facets of the EP program that involve actions by licensee staff are generally referred to as *onsite EP*. The EP program, procedures and systems maintained to implement governmental actions are referred to as *offsite EP*. The licensee is responsible for developing and implementing the onsite EP program and provides support to the offsite program as required. The NRC is responsible for assessing the adequacy of the overall program, but relies on the Federal Emergency Management Agency (FEMA) to assess the offsite program. The development and collection of performance indicators (PIs) for offsite EP by licensees is not considered necessary or appropriate at this time, because FEMA performs regular assessments of offsite EP programs.

The key attributes of an EP program are those program elements that are critical to achievement of the performance expectation. These key attributes are depicted in Figure 1 and are: ERO performance as demonstrated during simulated and actual events, Emergency Response

Figure 1



Organization (ERO) readiness to implement the Plan, the readiness of the facilities and equipment that support the ERO, the quality of procedures that support EP, and Offsite EP.

ERO Performance

The implementation of the Plan is dependant on the performance of the ERO in their EP assignments. The technical aspects of these assignments generally align with the expertise of the individual, but also include duties unique to EP. The opportunity to demonstrate proficiency is provided during drills, exercises and events that require implementation of the Plan. There are many areas important to Plan implementation, but the most risk significant areas of ERO performance are:

- <u>Timely and accurate classification of events</u>; including the recognition of events as potentially exceeding emergency action levels (EALs) and any assessment actions necessary to support classification; this is measured by a PI.

- <u>Timely and accurate notification of offsite governmental authorities</u>; including adequate performance of notifications as specified in the Plan; this is measured by a PI.

- <u>Timely and accurate development and communication of protective action recommendations to offsite authorities</u>; including providing appropriate protective action recommendations (PARs) to governmental authorities, the decision making process to develop PARs and accident assessment as necessary to support PAR development; this is measured by a PI.

ERO Readiness

Implementation of the Plan is dependant on the readiness of the ERO to respond to emergencies. Licensee training programs provide the ERO knowledge base, but drills and exercises provide opportunities to gain proficiency in individual duties and team functions in the integrated organization. Self assessment of performance during drills and exercises identifies successful performance and areas for improvement. Self assessment and corrective action resolution is critical to ERO proficiency. In this way, the drill program ensures a high level of ERO proficiency. In addition, timely ERO augmentation of on shift personnel is critical to overall performance. The most risk significant areas of ERO readiness are:

ERO Drill Participation; including participation in drills and other appropriate opportunities for proficiency development (and supplemented by self assessment of performance and corrective action-see below); this is measured by a PI.

ERO Performance Assessment: including self assessment of performance and identification of deficiencies in ERO performance, conduct of reviews required by 10 CFR 50.54 (t), identification of trends in deficiencies and the efficacy of the corrective action program; this is measured by inspection.

Timely ERO Augmentation; including the functioning of notification systems, adequacy of ERO response and adequacy of the duty roster to provide 24 hour staffing; this is measured by inspection.

Facilities and Equipment

Facilities and equipment required to implement licensee emergency response are specified in the Plan. The readiness of facilities and equipment to support ERO operations is a risk significant area:

- Availability of the Alert and Notification System; including the ability of the systems to perform their design function; this is measured by a PI.

- Availability of facilities and equipment; including surveillance of communications channels, facilities and equipment; this is measured by inspection.

Procedure Quality

Emergency Plan Implementing Procedures (EPIPs) are used by the ERO to implement the Plan. The response is tested regularly in drills and exercises and the quality of EPIPs is generally improved through assessment of performance. The emergency action levels (EALs) are delineated in the Plan and implemented through the use of an EPIP. Changes in EALs may be made by the licensee in accordance with 10 CFR 50.54 (q), but must be approved by NRC. Other procedures, such as severe accident management guides (SAMG) are also important to emergency response. The risk significant areas of procedure quality are:

- Classification of events; this is measured by a PI.

- Notification of offsite governmental authorities; this is measured by a PI.

- Development and communication of protective action recommendations to offsite authorities; this is measured by a PI.

- Configuration of the EALs; this is measured by inspection.

- Implementation of SAMG[4]; this is measured by inspection.

Offsite EP

State and local governmental authorities are responsible for maintaining the offsite EP programs and implementing protective actions to protect the public health and safety. While the licensee must supply appropriate information to governmental authorities to allow the timely implementation of protective actions, only governmental authorities are authorized to implement these actions. Implementation of offsite Plans is assessed in regular FEMA evaluations.

PERFORMANCE INDICATORS

Compliance of EP programs with regulation is assessed through observation of response to simulated emergencies and through routine inspection of onsite programs. Demonstration exercises involving onsite and offsite programs, form the key observational tool used to support, on a continuing basis, the reasonable assurance finding that *adequate protective measures can and will be taken in the event of a radiological emergency*. This is especially true for the most risk significant facets of the EP program. This being the case, the PIs proposed for onsite EP draw significantly from performance during simulated emergencies and actual declared emergencies, but are supplemented by direct NRC inspection and inspection of licensee self assessment. NRC assessment of the adequacy of offsite EP will rely (as it does currently) on regular FEMA evaluations.

1.0 **Drill, Exercise and Actual Event Performance (DEP)**; data collected quarterly, for use in a six month trend and a two year rolling average.

This PI consists of: The fraction, numerator and denominator, of successful performance actions over all opportunities for:

Classification of Emergencies

Notification

Protective Action Recommendation

Basis: Recognition and subsequent classification of events is a risk significant activity. Classification leads to augmentation of the ERO, as appropriate to the emergency class and notification of governmental authorities.

[4] The appropriateness of inspecting the implementation of SAMGs is under review as SAMGs are a voluntary industry initiative.

Timely and accurate notification of offsite authorities is a risk significant activity. Notification leads to mobilization of governmental authorities, as appropriate.

The timely and accurate development and communication of PARs is a risk significant activity. It requires that several supporting activities be performed including: assessment of plant conditions, quantification of radiological release magnitude, projection of the potential dose to the public and communication to government authorities. Communication of PARs leads to actions by governmental authorities to protect the public health and safety.

If the ERO consistently performs these activities in a timely and accurate manner, it indicates that the EP program is operating at or above the *threshold of licensee safety performance above which the NRC can allow licensees to address weaknesses with NRC oversight through a risk informed baseline inspection program.*

Requirements: Only activities that the licensee formally assesses for the timely and accurate performance of classification, notification and PAR development and communication, may be included in this statistic. Simulated emergency events that are identified as in advance of performance opportunities for this PI shall be included in the statistics, i.e., a candidate opportunity can not be removed from the data set due to poor performance. Opportunities shall include actual emergency declarations and the biennial exercise and may include other drills of appropriate scope and operating shift simulator evaluations conducted by the licensee training organization.

A drill may be considered of appropriate scope if it provides a proficiency development opportunity for the ERO that involves, or reasonably simulates the interaction as appropriate, of the control room, TSC, OSC, EOF, field monitoring teams, damage control teams and offsite governmental authorities, e.g., a field monitoring team may only interact with the EOF, the control room may only interact with the TSC, but the TSC may interact with the EOF, control room, OSC and government.

Operating shift simulator evaluations may be included only when the scope requires classification (if it were a real event) and notifications are performed at least to the point of filling out the appropriate forms and demonstrating sufficient knowledge to perform the actual notification. However, there is no intent to disrupt ongoing operator qualification programs. Appropriate operator training evolutions should be included in the statistics only when EP aspects are consistent with training goals.

There is no requirement to include any given drill or training evolution and no minimum is set for these observational opportunities. However, analyses performed on the data will recognize that a PI value generated by a greater number of opportunities more accurately represents licensee performance. Statistical opportunities should include multiple events during a single drill, evolution, etc., if supported by the scenario as follows:

- each expected recognition and classification opportunity should be included,

- notification opportunities should include notifications made to the state/local governmental authorities for initial emergency classification, upgrade of emergency class, initial PARs and changes in PARs, (periodic follow up notifications/briefings provided when classification or PARs have not changed are not included) and

- PAR opportunities should include the initial PAR and any appropriate PAR change.

Data Reporting Frequency

Data would be reported every 3 months

PI Threshold

The threshold for the white zone is two fold:

- < 75% for the previous six months

- < 90% for the previous two years

The first threshold, also referred to as “short-term performance threshold” is designed to trigger NRC action if the licensee’s performance declines over the past 6 months. The second threshold, also referred to as “long-term performance threshold” is designed to trigger NRC action upon licensee’s performance declines over the past 24 months. This dichotomy between “short-term” and “long-term” thresholds was deemed necessary to balance the significance of short-term and long-term performance indications (a decline in performance over a long period of time may be more significant than a sharp decline noted in a shorter period).

Basis for the 24-months Threshold

The long-term performance threshold of 90 % has been determined based on an analysis of emergency preparedness inspection findings from 1994 to 1997. The findings were extracted from inspection reports of NRC evaluated biennial exercises. A systematic assessment of each finding related to the risk-significant areas (classification, notification, PARs) was performed to determine successful performance and the number of opportunities. Successful performance was rolled up for each 24 month period and the corresponding average and standard deviation calculated. The following results were found:

24 months period	Number of Failures	Estimated Number of Opportunities *	Average of Successes	Standard Deviation

1996-1997	24	680	**96 %**	**9 %**
1994-1995	27	730	**96 %**	**7 %**

* 4 opportunities per exercise for Classification and 4 opportunities for Notification
2 opportunities per exercise for PARs

Examining Inspection reports was found to be a timely and convenient method to access past performance data similar to that which will be used for the DEP PI. However, inspection reports only address the opportunities associated with NRC evaluated exercises, which are a small part of the opportunities that licensees have to measure Drill/Exercise performance. Other licensee opportunities to measure performance stem from non-NRC evaluated drills, shift simulator evaluations, and actual emergency events. Data regarding those opportunities were not available for analysis to determine the DEP Threshold. However, the data extracted from NRC evaluated exercises is believed to represent a sample of typical licensee performance. In any given 24 month period, about 70 exercise inspections are performed by the NRC (68 were performed in the period 1996-1997 and 73 in the period 1994-1995), representing about 700 opportunities for classification, notification and PARs for the whole industry. On a plant-specific basis however, this corresponds to approximately 10 opportunities per 24 months. In the future, it is expected that licensees will collect and report annually 60 or more Drill/Exercise performance opportunities.

Based on the analysis of inspection findings, an emergency preparedness expert panel composed of NRC and industry representatives came to agreement on the value for the long-term threshold. The panel developed the threshold by taking the past 4 year average, diminishing it by one standard deviation, and rounding it up. The panel reached consensus with a 90 % long-term threshold.

Basis for the 6-months Threshold

Like the long-term threshold (24 months), the short-term threshold (6 months) has been selected based on the inspection findings analysis. The short-term threshold is less stringent that the long-term threshold to allow licensee to correct problems and yet, indicate trends. An emergency preparedness expert panel composed of NRC and industry representatives came to agreement on fixing this threshold at 75%.

Discussion of past individual plant performance against the thresholds

Analyses of individual plant performance against the 24 month threshold were performed for the period 1994-1997. Had PIs been in use in this period, they would have shown the following:

	Number of Plants not Meeting the Threshold	Total Number of Plants	Percent of Plants Meeting the Threshold
1994	1 (1)	70	98.5 %
1995	3 (2)	70	96 %

1996-1997	7 (3)	70	90 %

(1) Cooper
(2) Cooper, Palo Verde, Wolf Creek
(3) Haddam Neck, Three Mile Island, Prairie Island, Quad Cities, Palo Verde, River Bend, Washington Nuclear 2

This analysis confirms the reasonableness of the value chosen for the long-term threshold.

Threshold Limitations

In 1998, the NRC identified Clinton as a plant with a large number of concerns in emergency preparedness. However, The DEP PI for Clinton for the past 4 years indicates outstanding performance. Therefore, the indicator would not have identified Clinton weaknesses. In contrast, the DEP PI for Three Mile Island appropriately shows the decline in performance that was identified in 1997 during NRC inspections.

Tests of individual plant performance against the 6 month threshold have not been performed due to the lack of sufficient plant-specific data in any 6 month interval. It is believed that sufficient data will be available in the future to validate the short-term threshold. The 6 month threshold could be validated after a year of implementation. This will also give an opportunity to revisit the 24 month threshold.

Yellow Zone Thresholds

The threshold for the yellow zone is two fold:

- < 55% for the previous six months

- < 70% for the previous two years

Threshold Basis

An emergency preparedness expert panel composed of NRC and industry representatives came to agreement to utilize the 55% and 70% thresholds for the yellow zone of the DEP PI. There is some basis in the statistics for the 70% value in that 3σ below the industry average is about 73%. This was rounded down due to the severity of the Yellow Zone threshold. These values will be revisited after two years of implementation.

Implementation Proposals

For sake of first-time implementation, the long-term threshold tandem of 90/70% (green/red zone) should not be used until sufficient data have been gathered (24 months of data). In contrast, the short-term threshold tandem of 75/55% could be used after 6 months of data. If necessary for implementation, it could be assumed that in the 3 months prior to implementation, all licensees demonstrated 96 % successful performance, which corresponds to the past four years average. This may not be necessary if the PI based assessment program is not implemented until sufficient data is accumulated.

Verification

There are three aspects of PI data verification critical to this PI:

- The licensee self assessment program must accurately judge successful performance of classification, notification and PAR development and communication. This requires that the self assessment program be inspected to ensure veracity of data collection.
- Performance opportunities should be verified to be of sufficient depth to simulate ERO activation.
- The statistics of data collected from drills and other training evolutions should be verified periodically.

2.0 Emergency Response Organization Drill Participation (ERO)

This PI consists of: Percentage of ERO and operating shift crews that have participated in a drill, exercise or an actual event in the past 24 months and 36 months.

Basis: EP programs ensure the readiness of ERO personnel, facilities and equipment to support response to emergencies and protect the public health and safety. The licensee self assessment program is critical to ensuring readiness and does so through identification and correction of deficiencies. Drills and exercises tax the ERO, EPIPs and supporting facilities and equipment and self assessment of these events is a critical element of ensuring readiness.

The previous PI, DEP, measures the performance of segments of the ERO in risk significant activities during simulated and actual emergencies. However, the breadth and scope of the ERO activities include several important supporting areas not fully measured by DEP, such as accident assessment, dose projection, damage control, worker protection, and the ability to work as an integrated team under (simulated) emergency conditions. **ERO** measures opportunities that the total ERO has been given to gain proficiency as an integrated organization. It is expected that the licensee will assess these training opportunities to identify areas for improvement and that the corrective action program will ensure improvements are carried out. It is expected that these proficiency development opportunities will contribute to overall ERO readiness. In this way **ERO** indicates the proficiency of the ERO. **ERO** indirectly measures facilities and equipment readiness, training program efficacy and procedure quality because the licensee self assessment program can be expected to improve deficiencies in these areas uncovered in drills and exercises.

If a licensee consistently ensures that the ERO is proficient, it indicates that the EP program is operating at or above the *threshold of licensee safety performance above which the NRC can allow licensees to address weaknesses with NRC oversight through a risk informed baseline inspection program.*

Requirements: The ERO participation is intended to include the minimum positions committed to in the Plan and operating shift crews. This would include positions required for the functioning of the control room, TSC, OSC and EOF during Plan implementation. Plant workers, security personnel and others that are on shift or may be called in to support the emergency but do not fill positions on EP duty rosters or are not part of the operating shift crews, are not required to be included in this PI. However, positions that are formally on the EP duty roster, but not committed to in the Emergency Plan and others important to emergency response may be included.

Participation may be as a participant, mentor, coach, evaluator or controller (but not as an observer). Only participation in the drills, exercises and evolutions that are used to provide input to the DEP PI may be used in the statistics for this PI. Multiple assignees to a given ERO position could take credit for the same drill if their participation is a meaningful and thorough opportunity to gain proficiency in the assigned position.

Evaluated simulator evolutions that contribute to the DEP PI statistics could be considered for operation shift crew participation. However, there is no intent to disrupt ongoing operator qualification programs. Appropriate operator training evolutions should be included in the statistics only when EP aspects are consistent with training goals. If all crews have participated in more than one evaluated simulator evolution during the measurement period, it may only be counted as 100% and not more.

Data Reporting Frequency

Data would be provided every 3 months

PI Threshold

The threshold for the white zone is two fold:

- < 80% for the previous two years
- < 90% for the previous three years

Threshold Basis

No past data was readily available to help set the threshold value. An emergency preparedness expert panel composed of NRC and industry representatives came to an agreement to utilize the 80% and 90 % thresholds for ERO readiness. These values will be revisited after a year of implementation.

Yellow Zone Thresholds

The thresholds for the yellow zone are:

- < 60% for the previous two years
- < 70% for the previous three years

Threshold Basis

These thresholds were agreed upon between an NRC and industry representative emergency preparedness expert panel.

Implementation Proposals

The ERO readiness performance indicator and corresponding thresholds can be implemented immediately if the licensee has supporting data, otherwise at least two years of data would have to be collected before implementation.

Verification

Verification of the statistics used to generate the PI value is critical to this PI.

3.0 Alert and Notification System Availability (ANSA)

This PI consists of: Percent availability of Alert and Notification System

Basis: The Alert and Notification System (ANS) is a critical link for alerting and notifying the public of the need to take protective actions. Generally, the licensee maintains the ANS and state and/or local governmental authorities are responsible for activating it when necessary. Assurance that the system has a high rate of availability increases the assurance that the licensee can protect the public health and safety during an emergency.

If an EP program consistently ensures that the ANS is in a high state of readiness it indicates that the program is operating at or above the *threshold of licensee safety performance above which the NRC can allow licensees to address weaknesses with NRC oversight through a risk informed inspection program.*

Requirements: Statistical information gathered in support of system availability reports given to FEMA would form basis of this PI. It is proposed that the following rules be applied to gathering of this data:

- Failure of a siren is indicated by failure of any portion of the system that would have prevented it from performing its safety function, i.e., creating its design sound level and pattern.
- The period assumed for the failure would be in accordance with FEMA direction on gathering statistics.
- Periodic testing is in accordance with FEMA guidance and actually tests the ability of the siren to perform its intended safety function.

Data Reporting Frequency

Data would be provided every 3 months

PI Threshold

The threshold for the white zone is:

- < 94% for the previous year

Threshold Basis

The threshold of 94% has been determined based on an analysis of yearly sirens availability for 1995, 1996 and 1997 for approximately 20 plants. The two lowest values found were 91.1 % and 95.1%, the rest of the values were all above 96%. The sirens availability average for the 20 plants was 97.9%. An emergency preparedness expert panel composed of NRC and industry representatives came to an agreement to utilize a 94% threshold for sirens availability. This value will be revisited after a year of implementation.

Yellow Zone Threshold

The threshold for the yellow zone is:

- < 90% for the previous year

Threshold Basis

This threshold is based on the FEMA acceptance criteria for sirens availability. It has been agreed upon by an emergency preparedness expert panel composed of NRC and industry representatives. However, it should be noted that the FEMA acceptance criteria is based on a calendar year, while this PI is a rolling average.

Implementation Proposals

For sake of first-time implementation of the Alert and Notification System performance indicator and thresholds, it could be assumed that 6 months prior to implementation, all licensees demonstrated 97.9 % of sirens availability performance, which corresponds to the calculated past four years average for the plants sampled. Alternately, the PI may not be implemented until adequate data is available.

Verification

Verification of the statistics used to generate the PI value is critical to this PI.

INSPECTION AREAS

The inspection areas discussed below are necessary to ensure the licensee EP program is operating at or above the *threshold of licensee safety performance above which the NRC can allow licensees to address weaknesses with NRC oversight through a risk informed baseline inspection program.* These inspection elements represent the risk significant areas that are necessary to complement the proposed PI program for an EP program operating in the green zone.

- Verification the collection of PI statistics and that data gathering is in compliance with the guidelines described for PIs.

- Verify by observation of drills and the biennial exercise, the capability of the licensee self assessment program to assess performance during drills, exercises, actual declared events and operator simulator evaluations and provide an accurate assessment of successes and failures during performance opportunities.

- Verify that Alert and Notification System availability testing is in compliance with guidance.

- Inspect licensee ERO augmentation tests.

- Inspect and approve EAL changes as required by 10 CFR 50.54(q).

- Review the adequacy of the self assessment and corrective action program to correct areas requiring improvement to ensure the cornerstone objective continues to be met including:
    - ERO proficiency in general,
    - ERO ability to diagnose plant accident conditions, formulate mitigating actions and implement them under accident conditions,
    - readiness and quality of EP equipment and facilities,
    - direct interface with offsite authorities during exercises and drills, e.g., in the area of PAR communication and technical support,
    - adequacy of communication channel testing and timely correction of communication channel deficiencies,
    - audits conducted under 10 CFR Part 50.54 (t)
    - implementation of severe accident management guidance during drills, and
    - adequacy of worker protection during exercises and drills.

Table 1
EMERGENCY PREPAREDNESS KEY ATTRIBUTES AND MEANS TO MEASURE

Key Attribute	Areas to Measure	Means to Measure	Comments
ERO Performance	Timely and accurate classification of events	PI	Recognition and subsequent classification of events is a risk significant activity. Classification should lead to activation of the ERO as appropriate to the emergency class and notification of governmental authorities.
	Timely and accurate notification of offsite governmental authorities	PI	Timely and accurate notification of offsite authorities is a risk significant activity. Notification should lead to mobilization of governmental authorities, as appropriate.
	Timely and accurate development and communication of protective action recommendations to offsite authorities	PI	The timely and accurate development and communication of PARs is a risk significant activity. It requires that several supporting activities be performed in a timely manner to develop the PAR. Communication of PARs should lead to actions by governmental authorities to protect the public health and safety.

Key Attribute	Areas to Measure	Means to Measure	Comments
ERO Readiness	ERO Drill Participation	PI	**ERO** measures opportunities that the total ERO has been given to gain proficiency as an integrated organization. It is expected that the licensee will assess these training opportunities to identify areas for improvement and that the corrective action program will ensure improvements are carried out. It is expected that these proficiency development opportunities will contribute to overall ERO readiness. In this way **ERO** indicates the proficiency of the ERO.
	Demonstration of timely augmentation of the ERO	Inspection	Augmentation of on shift personnel with the ERO is critical to implementing the Plan in a timely manner during emergencies. This is a risk significant area of EP.
	Licensee self assessment	Inspection	Self assessment of ERO performance and identification of deficiencies is critical to maintaining an adequate level of ERO performance to give reasonable assurance. Conduct of reviews required by 10 CFR 50.54 (t) and the efficacy of the corrective action program to correct identified deficiencies and identify trends in deficiencies would be inspected.

Key Attribute	Areas to Measure	Means to Measure	Comments
Facilities and Equipment	Availability of the Alert and Notification System	PI	The Alert and Notification System (ANS) is a critical link for alerting and notifying the public of the need to take protective actions. Generally, the licensee maintains the ANS and state and/or local governmental authorities are responsible for activating it when necessary. Assurance that the system has a high rate of availability increases the assurance that the licensee can protect the public health and safety during an emergency.
	Availability of equipment and facilities	Inspection	Facilities, communications channels and equipment that are critical to the functioning of the ERO during emergencies must be maintained. The licensee self assessment program will address these areas and the inspection would evaluate this self assessment.
Procedure Quality	Classification of events	PI	EPIPs are used to classify events. The quality of the EPIPs will be reflected in the DEP PI as integral to the measured success rate.
	Notification of offsite governmental authorities	PI	EPIPs are used to notify governmental authorities. The quality of the EPIPs will be reflected in the DEP PI as integral to the measured success rate.

Key Attribute	Areas to Measure	Means to Measure	Comments
	Development and communication of protective action recommendations to offsite authorities	PI	EPIPs are used to develop and communicate PARs. The quality of the EPIPs will be reflected in the DEP PI as integral to the measured success rate.
	EAL changes are in accordance with 50.54 (q)	Inspection	Licensees may change the EAL set, but NRC must approve the change. This inspection element will review any changes against the requirements of 50.54 (q) and approve the change as appropriate.
	SAMG implementation	Inspection	SAMG drills are generally implemented by the ERO. Licensees will assess the effectiveness of that implementation and this inspection element will evaluate the self assessment.
Offsite EP	Implementation of State and local emergency plans	FEMA Evaluation	State/local governmental authorities are responsible for implementing emergency plans that include protective actions to protect the public health and safety. FEMA evaluates the effectiveness of implementation.

Table 2
Performance Indicators

PI Name	Measurement Area	Definition	Threshold
Drill/Exercise Performance (DEP)	Timely and accurate classification of events Timely and accurate notification of offsite governmental authorities Timely and accurate development and communication of protective action recommendations to offsite authorities	Fraction, (numerator and denominator,) of successful performance events over all opportunities for: <u>Classification of Emergencies</u> <u>Notification</u> <u>Protective Action Recommendation</u>	The threshold for the white zone is two fold: 75% for the previous six months 90% for the previous two years The threshold for the yellow zone is two fold: 55% for the previous six months 70% for the previous two years

PI Name	Measurement Area	Definition	Threshold
Emergency Response Organization Drill Participation (ERO)	Emergency Response Organization readiness	Percentage of ERO and operating shift crews that have participated in a drill or exercise in the past 24 and 36 months.	The threshold for the white zone is: 80% for the previous two years 90% for the previous three years The threshold for the yellow zone is: 60% for the previous two years 70% for the previous three years
Alert and Notification System Availability (ANSA)	Availability of the Alert and Notification System	Percent availability of Alert and Notification System	The threshold for the white zone is: 94 % for the previous year The threshold for the yellow zone is: 90% for the previous year

Appendix E
Occupational Exposure Cornerstone

General Description

This cornerstone includes the attributes and the bases for adequately protecting the health and safety of workers involved with exposure to radiation from licensed and unlicensed radioactive material during routine operations at civilian nuclear reactors. The desired result is the adequate protection of worker health and safety from this exposure. The cornerstone uses as its bases the occupational dose limits specified in 10 CFR 20 Subpart C and the operating principle of maintaining worker exposure "as low as reasonably achievable (ALARA)" in accordance with 10 CFR 20.1101. These radiation protection criteria are based upon the assumptions that a linear relationship, without threshold, exists between dose and the probability of stochastic health effects (radiological risk); the severity of each type of stochastic health effect is independent of dose; and nonstochastic radiation-induced health effects can be prevented by limiting exposures below thresholds for their induction. Thus, 10 CFR Part 20 requires occupational doses to be maintained ALARA with the exposure limits defined in 10 CFR 20 Subpart C constituting the maximum allowable radiological risk. Industry experience has shown that the occurrences of uncontrolled occupational exposure which potentially could result in an individual exceeding a dose limit have been low frequency events. These potential overexposure incidents are associated with radiation fields exceeding 1000 millirem per hour (mrem/hr) and have involved the loss of one or more radiation protection controls (barriers) established to manage and control worker exposure. The probability of undesirable health effects to workers can be maintained within acceptable levels by controlling occupational exposures to radiation and radioactive materials to prevent regulatory overexposures and by implementing an aggressive and effective ALARA program to monitor, control and minimize worker dose.

Occupational Exposure Key Attributes

Those attributes which affect worker exposure at an operating facility are shown in Figure 1. These attributes affect the licensee's ability to control individual worker exposures and, furthermore, to maintain occupational exposures ALARA. The control of occupational exposure can be maintained at an acceptable level by minimizing human performance errors, implementing quality programs and processes, and assuring the proper design and use of plant equipment and instrumentation for radiation protection activities. Acceptable performance within these key attributes would result in a low frequency of significant occupational exposure events which could result in a regulatory limit being exceeded or in the ineffective implementation of ALARA program resulting in unnecessary occupational exposure.

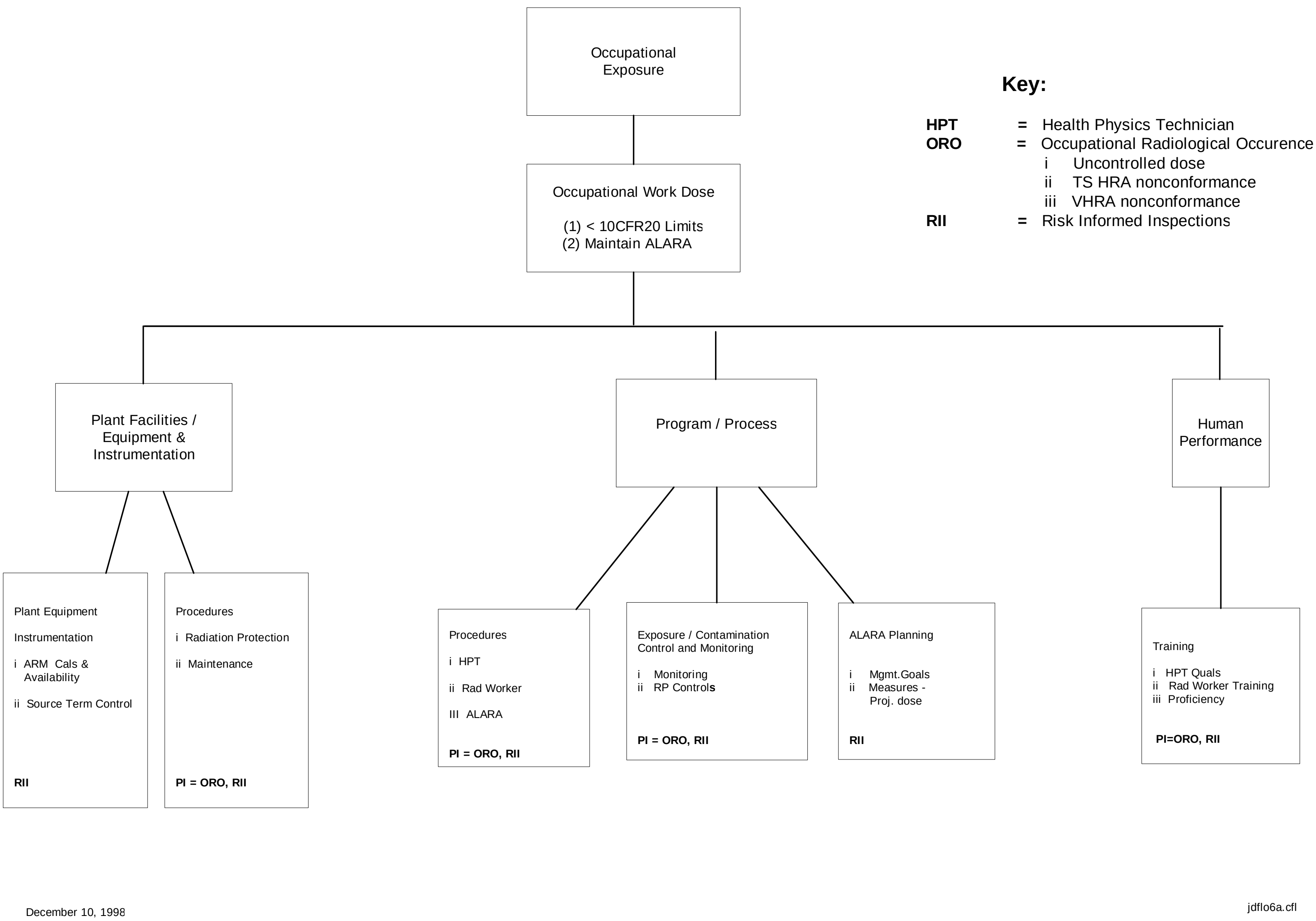


Figure 1

Within each occupational exposure cornerstone attribute, specific areas for measurements either by performance indicators or by inspection activities have been identified for assessment.

Facilities, Equipment, Instrumentation

Inoperable monitoring instrumentation and inadequate source term control can result in significant unplanned exposures. For selected facility areas, e.g., Boiling Water Reactor (BWR) Transverse Incore Probe (TIP) drive room, reliable and accurate area radiation monitors (ARMs) can remotely identify transient high dose rate fields to reduce the potential for uncontrolled exposure. In addition, the use of chemical decontamination processes and the proper design and installation of shielding associated with equipment and systems having elevated source terms can be used to effectively reduce the potential for uncontrolled or unnecessary occupational exposures . The effectiveness of this key area in meeting the cornerstone objective is dependent upon acceptable radiation protection procedures for source term evaluation and reduction and for maintenance and calibration of radiation protection systems and equipment. These measurement areas are considered to be site specific and more appropriately assessed through the baseline inspection process.

Program/Processes

The technical adequacy of radiation protection procedures and proper implementation of program processes contribute to the control and minimization of occupational exposures. Improper radiological surveillances have resulted in significant uncontrolled occupational exposure from direct exposure to radiation sources or from intakes of radioactive material. The establishment of administrative and physical radiation protection controls serve as additional mechanisms (barriers) preventing uncontrolled worker access to high radiation, significantly contaminated and airborne areas. The development of aggressive dose expenditure goals, combined with detailed work planning, accurate assessment of associated radiological conditions and establishment of adequate controls are necessary to implement an effective ALARA program. In particular, these planning, monitoring and control activities increase in importance during outages when interfaces between personnel and high radiation and contaminated systems increase. Within this cornerstone attribute, the PI can monitor the performance in a limited number of measurement areas with the majority of assessment requiring direct baseline inspection effort.

Human Performance

Human performance can significantly affect occupational worker exposures during work activities conducted in elevated dose rate and contaminated areas. Inadequate performance by health physics technicians (HPTs) or workers can result in a loss of the multiple radiation protection barriers established to prevent uncontrolled exposures. In addition, adherence to proper radiation protection practices is necessary to implement an effective ALARA program. A combination of PI information and inspection is proposed to properly assess this area.

Performance Indicators

A combined performance indicator (PI) is proposed to assess licensee performance in controlling worker doses during work activities associated with high radiation fields or elevated airborne radioactivity areas. The PI was selected based upon its ability to provide an objective measure of an uncontrolled measurable worker exposure or a loss of access controls for areas having radiation fields exceeding 1000 millirem per hour (mrem/hr). The data for the PI are currently being collected by most licensees in their corrective action programs. The PI either directly measures the occurrence of unanticipated and uncontrolled dose exceeding a percentage of the regulatory limits or identifies the failure of barriers established to prevent unauthorized entry into those areas having dose rates exceeding 1000 mrem/hr. The indicator may identify declining performance in procedural guidance, training, radiological monitoring, and in exposure and contamination control prior to exceeding a regulatory dose limit (Table 1). The effectiveness of the licensee's assessment and corrective action program is considered a cross-cutting issue and is addressed elsewhere. The three components of the occupational radiological occurrence (ORO) PI are defined as follows:

Occupational Radiological Occurrence

1a. **Technical Specification High Radiation Area (TS HRA) Occurrence:** A single nonconformance with TS controls or comparable 10 CFR Part 20 requirements applied to high-radiation areas (HRAs) with dose rates greater than or equal to ($\geq$) 1000 millirem per hour (mrem/hr). Where licensee TSs do not address controls for HRAs $\geq$ 1000 mrem/hr, nonconformance with comparable provisions in licensee procedures will define an occurrence.

1b. **Very High Radiation Area (VHRA) Occurrence:** A single nonconformance with 10 CFR Part 20 and/or licensee procedural requirements regarding radiation protection controls, i.e., postings, area surveys, personnel monitoring, administrative controls and physical barriers associated with VHRAs, areas having radiation fields $\geq$ 500 rad/hr at one meter.

1c. **Uncontrolled Exposure Occurrence:** A single occurrence resulting in one or more uncontrolled occupational exposures equal to or exceeding ($\geq$) 10 percent (%) of the 10 CFR Part 20 non-stochastic and/or 2% of the stochastic limits specified in 10 CFR Part 20. For minors and declared pregnant women, an uncontrolled exposure occurrence will be defined as doses >20 % of the stochastic limits detailed in 10 CFR 20.1207 and 10 CFR 20.1208. For skin exposure from "hot particles," an occurrence will be defined as exposures $\geq$ 100 % of the current established limit.

Calculational Method: Rolling sum of PI components 1a through 1c based on either the previous 36 month (long-term) or previous 12 month (short-term) interval. Data to be collected quarterly (every three months).

Thresholds

Increased Regulatory Oversight (Green-White): Six or more occupational radiological occurrences, summation of 1a through 1c, within a rolling three-year interval or three or more occurrences within a rolling twelve-month interval.

The preliminary short and long-term thresholds are based on a review and analysis of quarterly occupational radiological occurrence data provided by 28 licensee sites for the period January 1996 through September 1998 (i.e., 11 quarters or 2.75 years). From analysis of the data provided, a long-term average of approximately 1.3 occurrences per site within the three-year interval was calculated. Based on the mean and the associated standard deviation, approximately 95 percent of the sites were expected to have five OROs within a three year interval. An expert panel composed of NRC and industry representatives agreed to utilize six or more occurrences within a three year interval for the preliminary long-term threshold. The short-term threshold was established at 50 percent of the long-term threshold, i.e., three (3) or more occurrences within a year.

Sites exceeding either the long or short-term PI thresholds were compared against those sites with performance in occupational radiation protection activities identified by NRC regional staff as not meeting or declining from industry standards. For the 12 identified sites, data were not available for five sites, three sites exceeded either the short or long-term thresholds, two sites trended above the long-term average of 1.3 OROs per three- year interval, and two sites did not exceed the threshold and were not above the average ORO three-year average. Although, two sites not identified by the NRC staff as not meeting or having declining performance exceeded the PI thresholds; the identified OROs transpired early in the three-year interval and prior to the current SALP cycle. Excluding one site, all the facilities exceeding the preliminary thresholds were rated as SALP Category 2 or 3 in the plant support area. For the one plant support SALP Category 1 site which exceeded the threshold, the identified occupational radiological occurrences were prior to the current SALP cycle. None of the other sites ranked by NRC staff as meeting the industry norm or as a SALP Category 1 in plant support exceeded the preliminary threshold data. The alignment of NRC staff and plant support SALP ratings with the performance indicator thresholds advances their initial use in assessing licensee performance. Additional data are being collected to enhance the analysis and verification of the ORO performance indicator.

(White - Yellow) Twelve or more occupational radiological occurrences, summation of 1a through 1c, within a rolling three-year interval or six or more OROs within a rolling twelve-month interval.

An occupation exposure expert panel agreed to establish a preliminary threshold for the white-yellow zone as double both the short- and long-term threshold criteria. As expected, no sites which provided the initial ORO data exceeded the established threshold. These values will be reviewed subsequent to accumulation of additional data and a year of program implementation.

Inspection Areas

The purpose of the PI is to allow the inference to be made that, if the PI is below the threshold, then performance within that key attribute is appropriately monitored and controlled by the licensee. From an analysis of the industry experience, the proposed PI alone will not reflect licensee performance within each of the key attributes. Licensee performance in radiologically risk-important areas not covered by the PI will be assessed through a baseline inspection program. Table 2 details areas where PI and inspection are required for proper assessment of the cornerstone.

Facilities, Equipment, Instrumentation

Licensee performance associated with the identified measurement areas in this key attribute are site specific and should be evaluated as a component of baseline inspections. For example, area radiation monitors used to identify potentially significant transient high dose rate areas vary among licensee sites, and the availability and operability of those monitors should be assessed through baseline inspection. Source reduction methods, e.g., shut-down chemistry and the use of shielding to reduce potential high dose fields during outage operations, are also site specific and should be reviewed as part of the base-line inspection activities.

Program/Processes

The proposed PI is expected to measure licensee performance only in controlling areas with dose rates exceeding 1000 mrem/hr or where a significant uncontrolled dose to an individual results from either external radiation sources or from internally deposited radioactive material. Assessment of source term monitoring and reduction are site specific and highly dependent upon previous operational history, current work scope, and worker experience. Assessment of the ALARA activities will require direct inspection to evaluate licensee performance by comparing established goals to actual dose expenditures, with the established goals benchmarked to previous performance by the licensee.

Human Performance

The PI may not identify all instances of degraded performance in this area. For example, a recent failure of a health physics technician to assess radiological conditions and implement proper radiation protection controls for workers resulted from an improper evaluation of an alarm from a worker's electronic dosimeter. Direct inspection was necessary to determine the root cause of the degraded performance. Baseline inspection will assess the proficiency of health physics technicians in covering high dose rate (but less than 1000 mrem/hr) and high collective dose tasks and the proficiency of workers involved in such tasks.

<table>
<tr><th colspan="4">TABLE 1: OCCUPATIONAL EXPOSURE PERFORMANCE INDICATORS</th></tr>
<tr><th>PI</th><th>Measurement Area</th><th>Definition</th><th>Thresholds</th></tr>
<tr><td rowspan="3">Uncontrolled Exposure Occurrence
$\geq$ 100 mrem

or

Technical Specification High Radiation Area (TS HRA) Nonconformance

or

Very High Radiation Area (VHRA) Nonconformance</td><td>Procedures; Exposure and contamination monitoring and control; Training</td><td>A single occurrence resulting in one or more uncontrolled occupational exposures in excess of 10% of the non-stochastic and/or 2% of the stochastic limits specified in 10 CFR Part 20[1].</td><td rowspan="3">(Green -White) Increased Regulatory Oversight: Six or more occupational radiological occurrences within a rolling three-year interval or three or more occurrences within a rolling twelve-month interval.

(White - Yellow) Twelve or more occupational radiological occurrences within a rolling three-year interval or six or more occurrences within a rolling twelve-month interval.

White-Red: None Proposed</td></tr>
<tr><td>Procedures; Exposure and contamination monitoring and control; Training</td><td>A single TS or 10 CFR Part 20 nonconformance for HRAs with dose rates $\geq$ 1000 mrem/hr. Where licensee TSs do not address HRA controls, procedural nonconformance will define the PI.</td></tr>
<tr><td>Procedures; Exposure and contamination monitoring and control; Training</td><td>A single nonconformance with 10 CFR Part 20 and/or licensee procedural requirements regarding VHRA controls.</td></tr>
</table>

[1] For minors and declared pregnant women, an uncontrolled exposure occurrence will be defined as doses equal to or exceeding ($\geq$)20% of the stochastic limits detailed in 10 CFR 20.1207 and 10 CFR 20.1208. For skin exposure from “hot particles,” an occurrence will be defined as doses $\geq$ 100 % of the current established limit.

TABLE 2: OCCUPATIONAL EXPOSURE KEY ATTRIBUTES AND MEANS TO MEASURE

Key Attribute	Areas To Measure	Means to Measurement	Comments
Plant Facilities/Equipment /Instrumentation	Source Term Monitoring	Baseline Inspection	Review licensee programs for identifying and properly monitoring source terms resulting in high radiation areas (HRAs). For transient high dose rate areas, e.g., resin transfer operations, incore drive manipulation, and primary coolant leakage verify operability of select area radiation monitors (ARMs). Source term monitoring would be site specific and would be assessed by baseline inspection.
	Source Term Reduction	Baseline Inspection	Review licensee programs for source term control. Include plant modifications, shielding, and chemical decontamination activities. Program activities are site specific and would be assessed by baseline inspection.
	Procedures	Performance Indicator (PI); Baseline Inspection	Review licensee procedures for identifying and reducing elevated radiation field source terms. Evaluate the adequacy of procedures for maintaining and calibrating area radiation monitors.
Program/Process	Guidance /Procedures	PI; Baseline Inspection	The PI assesses performance in HRAs with dose rates $\geq$ 1000 mrem per hour (mrem/hr) and in VHRAs. For HRAs between 100 - 1000 mrem/hr baseline inspection would assess performance.
	Exposure & Contamination Monitoring/Control	PI Baseline Inspection	The PI assesses performance in HRAs $\geq$ 1000 mrem/hr and VHRAs. For HRAs between 100 - 1000 mrem/hr baseline inspection would assess performance.

TABLE 2: OCCUPATIONAL EXPOSURE KEY ATTRIBUTES AND MEANS TO MEASURE

Key Attribute	Areas To Measure	Means to Measurement	Comments
	ALARA Planning	Baseline Inspection	Inspection of ALARA program performance is site specific and will be assessed through baseline inspection. Assessment should be benchmarked against previous history.
Human Performance	HPT Qualifications & Performance	Inspection & PI	Baseline inspection to assess proficiency of HPTs covering high dose rate and high collective dose tasks. The PI may indicate degraded performance in HRAs $\geq$ 1000 mrem/hr and VHRAs.
	Radiation Worker Training/ Performance	Inspection & PI	Baseline inspection to evaluate proficiency of workers involved in high dose rate and high collective dose tasks. The PI may assess degraded performance in HRAs $\geq$ 1000 mrem/hr and VHRAs.

Appendix F
Public Exposure Cornerstone

General Description

This cornerstone includes the attributes and the bases for adequately protecting public health and safety from exposure to radioactive material released into the public domain as a result of routine civilian nuclear reactor operations. The desired result is the adequate protection of public health and safety from this exposure. These releases include routine gaseous and liquid radioactive effluent discharges, the inadvertent release of solid contaminated materials, and the offsite transport of radioactive materials and wastes. The cornerstone uses as its bases, the dose limits for individual members of the public specified in 10 CFR 20, Subpart D; design objectives detailed in Appendix I to 10 CFR Part 50 which defines what doses to members of the public from effluent releases are "as low as reasonably achievable" (ALARA); and the exposure and contamination limits for transportation activities detailed in 10 CFR Part 71 and associated Department of Transportation (DOT) regulations. These radiation protection standards require doses to the public be maintained ALARA with the regulatory limits constituting the maximum allowable radiological risk based on the linear relationship between dose received and the probability of adverse health effects.

Public Exposure Key Attributes

Licensee performance attributes that affect public exposure resulting from routine operations at a licensed facility are shown in Figure 1. The attributes affect a licensee's ability to accurately monitor and effectively control doses to members of the public from either direct exposure or release of radioactive materials into the public domain. Licensees can control and accurately measure or estimate doses to members of the public by sustaining acceptable human performance; ensuring the quality of established programs and processes; and optimizing reliability and accuracy of radioactive effluent processing and monitoring equipment. Acceptable performance within these key areas is part of a defense-in-depth strategy corroborating that doses to members of the public from routine civilian nuclear reactor operations are within established limits and are maintained ALARA. The performance indicator (PI) selected for this cornerstone assess the licensee's ability to effectively control and minimize, and accurately monitor radiation exposure to members of the public from routine operations. Within each attribute of the public exposure cornerstone, both performance indicator information and baseline inspection are required to assess licensee performance.

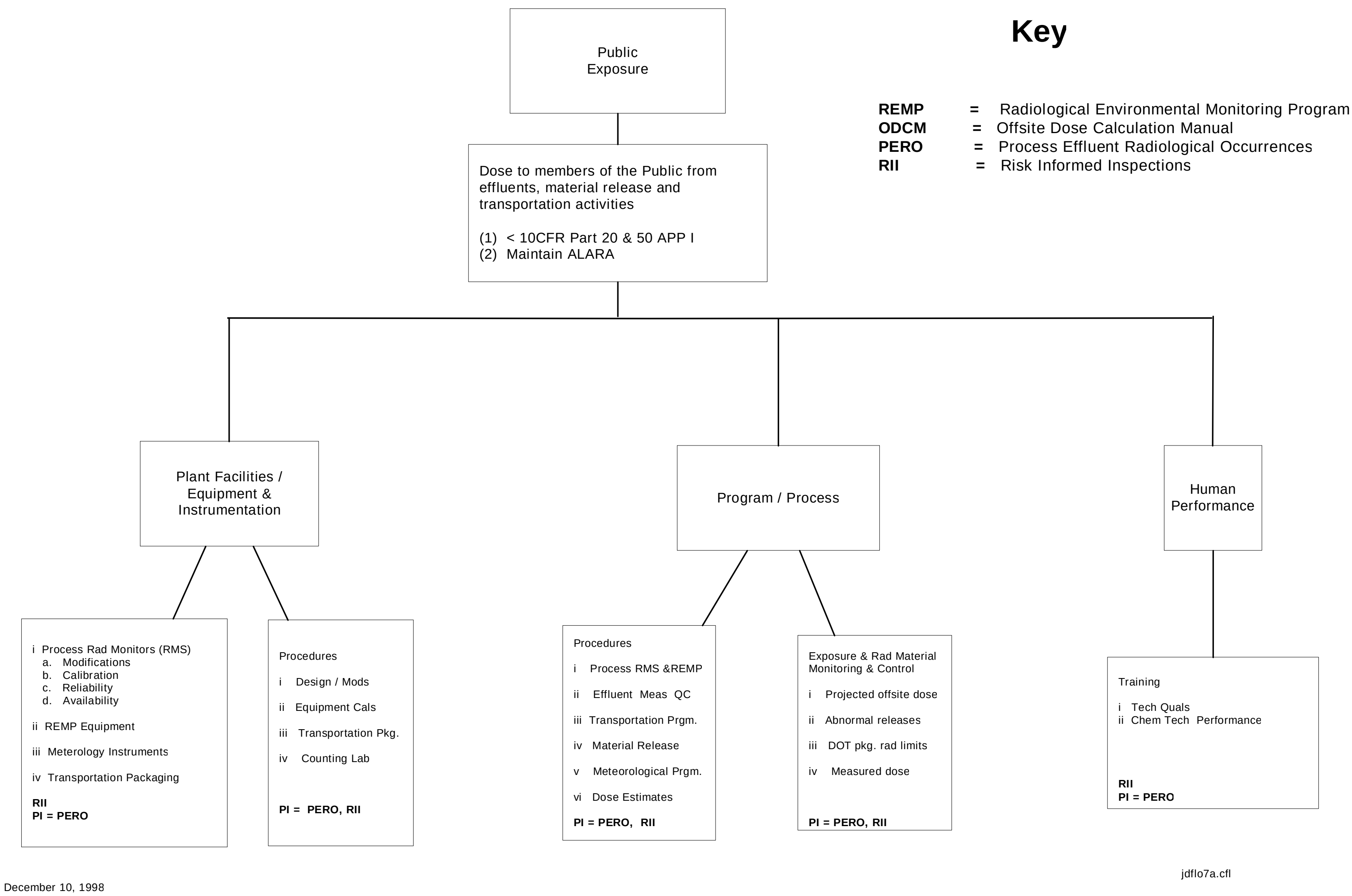


Figure 1

Facilities, Equipment and Instrumentation

Improper installation or modification, inaccurate calibration and reduced availability of meteorological systems, process radiation monitoring system (RMS) detectors and sampling systems, and associated counting room equipment adversely affect licensee performance in achieving and demonstrating compliance with effluent regulatory limits and design objectives. Similar issues affect the effectiveness of the radiological environmental monitoring program (REMP) equipment. The performance of radioactive waste (radwaste) processing, effluent sampling and monitoring equipment and instrumentation can be assessed, in part, by projected and measured offsite doses, and by RMS operability and availability. For transportation activities, shipping packages not prepared in accordance with their applicable design requirements, e.g., with the appropriate Certificate of Compliance (CoC) specifications for Type B shipments, increase the potential for unexpected exposure or loss of radioactive material which could result in uncontrolled and unnecessary exposures to members of the general public. The unconditional release of materials from protected areas requires the use of sensitive radiation survey equipment properly setup and calibrated to demonstrate the absence of significant contamination which could result in unnecessary dose to members of the public. Technically adequate procedures must be available for the meteorological and radiation systems design, modification, and calibration, for transport package preparation, and for counting room instrumentation setup and calibration.

Program/Process

Procedures must be technically adequate and implemented appropriately to conduct proper radiological effluent processing, and effective control and accurate monitoring of subsequent liquid and gaseous releases. Adequate procedures for routine system operation are required to ensure acceptable performance of meteorological instrumentation, radwaste processing, and process RMS equipment. For transportation activities, procedural guidance is necessary for proper evaluation of radwaste and material radionuclide quantities and types, for the subsequent selection and preparation of shipping packages and for conducting surveys to ensure that package radiological doses and contamination levels are within regulatory limits. The performance of radiological surveys for the unconditional release of potentially contaminated materials from licensee protected areas requires appropriate policy and technical guidance for handling and processing a wide variety of potentially contaminated materials. The PI will allow assessment, in part, of procedures and guidance for radwaste processing, effluent RMS operation, and for the survey and release of potentially contaminated materials outside of licensee’s protected areas.

Human Performance

Human performance can directly affect radwaste processing, effluent monitoring, and transportation activities. Human errors have contributed to incorrect release of radwaste tanks, inaccurate determination of RMS set points, and to abnormal and unmonitored effluent releases to the surrounding environs. In addition, health physics technician errors in conducting radiation surveys have contributed to shipping container dose rates or contamination levels exceeding regulatory limits or to the improper unconditional release of contaminated solid materials into the public domain. The identified PI will be combined with baseline inspection activities to assess human performance.

Performance Indicators

One PI for the radioactive effluent release program has been initially developed to monitor for inaccurate or increasing projected offsite doses (Table 1). The effluent radiological occurrence (ERO) PI does not evaluate performance of the radiological environmental monitoring program (REMP) which will be assessed through the routine baseline inspection. For transportation activities, the infrequent occurrences of elevated radiation or contamination limits in the public domain from this measurement area precluded identification of a corresponding indicator. A second PI has been proposed for future use to monitor the inadvertent release of potentially contaminated materials which could result in a measurable dose to a member of the public. These indicators will provide partial assessments of licensee radioactive effluent monitoring and offsite material release activities and were selected to identify decreasing performance prior to exceeding public regulatory dose limits.

Public Radiation Exposure Performance Indicators

1. **Process Effluent Radiological Occurrences:** Nonconformance with Radiological Effluent Technical Specifications (RETS) pursuant to radiological effluent releases; and excluding abnormal releases and out of service monitors, radioactive effluent release attributes reportable to the NRC in accordance with 10 CFR Part 20, Appendix I to 10 CFR Part 50, and Offsite Dose Calculation Manual (ODCM). For licensees having RETS removed from the Technical Specifications, nonconformance with comparable ODCM provisions.

   Calculational Method: Rolling sum of process effluent radiological occurrences (EROs) based on either the previous 36 month (long-term) or previous 12 month (short-term) interval. Data are to be collected quarterly (every three months).

   Thresholds:

   (Green-White Threshold) Increased Regulatory Oversight: Seven or more EROs within a rolling three-year interval or four or more EROs within a rolling twelve-month interval.

   The preliminary short- and long-term thresholds were based on a review and graphical analysis of Licensee Event Report (LER) data associated with process RMS activities provided by all sites for the period from January 1995 through December 1997, i.e. three years. Based on a graphical plot of the plant LER frequency data, approximately five percent of the sites had seven or more LERs during the period reviewed. An expert panel composed of NRC and industry representatives agreed to utilize seven or more occurrences within a three year interval for the preliminary long-term threshold. The short-term threshold was proposed as four or more occurrences within a rolling 12 month interval.

   Sites exceeding either the short- or long-term PI threshold were compared against those sites with performance in effluent measurements activities identified by NRC regional staff as not meeting or declining from industry standards. Of 12 sites identified by regional staff as performing below industry standards, four facilities exceeded either the short or long-term ERO PI thresholds. From subsequent review and discussion of the effluent monitoring LERs, the expert panel verified that not all events reportable to the NRC in accordance with the ODCM or RETS were included in the data submitted for benchmarking. Following receipt of additional reportable data from semiannual effluent reports these threshold values will be reviewed further.

   (White - Yellow Threshold) Fourteen or more EROs within a rolling three-year interval or eight or more occurrences within a rolling twelve-month interval.

An public exposure expert panel agreed to establish a preliminary threshold for the white-yellow zone as double both the short- and long-term threshold criteria. As expected, no sites which provided the initial ERO data exceeded the established threshold. These values will be reviewed subsequent to accumulation of additional data and a year of program implementation.

(Yellow - Red Threshold) None Proposed

2. **Unauthorized Radioactive Material Release:** Release of radioactive material(s) from licensee control which could reasonable result in public exposure in excess of 1 millirem per year (mrem/yr) total effective dose equivalent (TEDE).

   This PI is proposed for future use to assess licensee performance in effectively monitoring and preventing measurable dose to members of the public from the unconditional release of solid materials from the licensee protected area. The PI will be implemented subsequent to development of dose assessment methodology.

   Calculational Method: To Be Determined (TBD)

   Thresholds: TBD

Inspection Areas

For cornerstone measurement areas assuring and maintaining a defense in depth strategy regarding public health and safety which are not amenable to monitoring by the proposed PI, performance will be assessed through baseline inspection. For example, improper process RMS equipment designs and calibrations may not be identified by the currently proposed PI and would be reviewed as part of baseline inspection. For transportation and REMP activities, assessments will be conducted through the baseline inspection program. Cornerstone attribute measurement areas and their relationship to assessment by baseline inspection or PI are detailed in Table 2. All inspections are to be risk-informed or based on assessments of systems or processes necessary to maintain a defense-in-depth strategy regarding unexpected or unnecessary radiation levels or radioactive contamination within the public domain.

Facilities, Equipment and Instrumentation

The PI monitors performance in processing and monitoring radioactive effluents discharged into unrestricted areas. However, the proposed PI can only monitor acceptable performance assuming the systems and equipment (meteorological and RMS detectors and samplers) used in the radioactive effluent release and offsite dose assessments are installed, maintained and calibrated accurately, and the associated programs are implemented effectively. Baseline inspection are necessary to determine the adequacy of design modifications and calibrations of radioactive waste processing equipment and effluent monitoring instrumentation.

For transportation and REMP measurement areas, performance will be assessed through baseline inspection. In the transportation area, packaging for Type B radwaste or radioactive material shipments in accordance with the applicable certificate of compliance should be verified. For the REMP measurement area, inspection will include verification of sampling equipment location and operability.

Program/Process

The effectiveness of the radioactive waste processing and effluent monitoring, and transportation activities are dependent on technical adequacy of, and the proper implementation of procedures. Inadequate procedures have resulted in improper effluent release setpoints and or incorrect determination of effluent radionuclide concentrations. Within the transportation area, the determination of radionuclide types and quantities are dependent on technically adequate site specific procedures. Also, monitoring for the unconditional release of potentially contaminated materials from the protected area is dependent on consistent policy and technically adequate procedures. The PI does not evaluate situations were monitors were out-of-service or evaluation of abnormal releases. Inspection activities should verify acceptability of licensee action for these situations. Inspection should verify completion of QC sample analyses and acceptability of results for effluent measurements. In addition, inspection should verify the licensee program and methods for identifying radionuclides and quantities in shipments and for preparing marking, labeling and placarding for all packaging.

Human Performance

Human errors can significantly affect performance within the public exposure cornerstone. Inadequate performance of radiological surveys have contributed to instances of the unintentional release of licensed radioactive material or to transportation package dose and contamination levels exceeding regulatory limits. Human performance errors also have resulted in the release of radioactive waste from incorrect waste tanks, missed compensatory samples for out-of-service monitors and improperly calibrated detectors, These measurement areas will be assessed through baseline inspection to verify completion of applicable Hazardous Material Training requirements for all personnel involved in processing and loading packages of radioactive materials for transportation. Baseline inspection will also verify qualifications, training, and proficiency of health physics and chemistry technicians and radwaste operations staff involved in effluent processing.

TABLE 1: PUBLIC EXPOSURE PERFORMANCE INDICATOR

PI	Measurement Area	Definition	Thresholds
Process Effluent Radiological Occurrence	Offsite Dose Process Radiation Monitor	Nonconformance with Radiological Effluent Technical Specifications (RETS) pursuant to radiological effluent releases; and excluding abnormal releases and out of service monitors, radioactive effluent release attributes reportable to the NRC in accordance with 10 CFR Part 20, Appendix I to 10 CFR Part 50, and Offsite Dose Calculation Manual (ODCM). For licensees having RETS removed from the Technical Specifications, nonconformance with comparable ODCM provisions.	(Green-White Threshold) Increased Regulatory Oversight: Seven or more effluent radiological occurrence (EROs) within a rolling three-year interval or four or more occurrences within a rolling twelve-month interval. (White - Yellow Threshold) Fourteen or more EROs within a rolling three-year interval or eight or more occurrences within a rolling twelve-month interval. (Yellow - Red Threshold) None
Unauthorized Radioactive Material Release Occurrence **(Proposed - To Be Developed)**	Radioactive Exposure / Material Controls	Unauthorized release of radioactive material from the protected area which could reasonably result in a member of the public exceeding 1 mrem/yr TEDE	(TBD)

TABLE 2: PUBLIC EXPOSURE KEY ATTRIBUTES AND MEANS TO MEASURE

Key Attribute	Areas To Measure	Means to Measure	Comments
Plant Facilities, Equipment & Instrumentation	Process Radiation Monitoring System (RMS) Radiological Environmental Monitoring Program (REMP) Configuration Controls: Design & Installation	Baseline (BL) Inspection Performance Indicator (PI)	The PI monitors performance in processing and monitoring radioactive effluents discharged into unrestricted areas. Regulatory Guide 1.109 defines those significant release pathways to be monitored. The proposed PI monitors acceptable performance assuming correct system installation or modifications. Baseline inspection to be conducted on modifications of radioactive waste processing equipment and effluent monitoring instrumentation.
	RMS and Counting Room Detector Calibrations	Baseline Inspection	Baseline inspection to be conducted of calibration of process RMS detectors and counting room laboratory instrumentation associated with effluent monitoring activities.
	Meteorological Monitoring	Baseline Inspection	Baseline inspection verifies operability of meteorological instrumentation
	Transportation Packaging Configuration	Baseline Inspection	Site specific review of transportation activities required. Inspection to verify proper packaging for representative shipments. For Type B Shipments, verify implementation of Certificate of Compliance requirements.

TABLE 2: PUBLIC EXPOSURE KEY ATTRIBUTES AND MEANS TO MEASURE

Key Attribute	Areas To Measure	Means to Measure	Comments
	Procedures	PI Baseline Inspection	Review licensee procedures for maintaining, modifying, and calibrating meteorological and RMS equipment. Evaluate procedures for setup and evaluation of counting room equipment and for preparation of transport packages.
Program/Process	Projected Dose	PI	Verifies that radwaste processing and subsequent releases meet regulatory limits and are ALARA
	Radwaste Processing Effluent Monitor Operations	PI Baseline Inspection	For normal operations, PI establishes that releases are acceptable. The PI does not evaluate situations were monitors were out of service or evaluation of abnormal releases. Inspection activities should verify acceptability of licensee action for these situations.
	Radioactive Material Control: Inadvertent Release	PI	The PI assesses licensee ability to monitor and control releases of radioactive material and contamination which could result in a measurable dose to a member of the public.
	Effluent Measurement Quality Control	Baseline Inspection	Inspection verifies completion of QC sample analyses and acceptability of results for effluent measurements
	Transportation: DOT requirements	Baseline Inspection	Verify licensee program and methods for identifying radionuclides and quantities in shipments and for preparing marking, labeling and placarding for all packaging.

TABLE 2: PUBLIC EXPOSURE KEY ATTRIBUTES AND MEANS TO MEASURE

Key Attribute	Areas To Measure	Means to Measure	Comments
Human Performance	Technician / Operations Qualifications	Baseline Inspection	Verify qualifications and training of health physics or chemistry technicians involved in effluent processing. Verify training of radwaste operations staff. Effluent release occurrence PI may indicated degraded performance in this measurement area.
	HazMat Training	Baseline Inspection	Verify completion of applicable Hazardous Material Training requirements for all personnel, i.e., HP, Chem, Operations and Maintenance, involved in processing and loading packages of radioactive materials for transportation.
	Proficiency	Baseline Inspection	Observe HP, Chem and Operations staff proficiency in conducting radioactive processing and release activities.

Appendix G
Physical Security Cornerstone

General Description

This cornerstone addresses the attributes and establishes the basis to provide assurance that the physical protection system can protect against the design basis threat of radiological sabotage as defined in 10 CFR 73.1(a). The key attributes in this cornerstone are based on the defense in depth concept and are intended to provide protection against both external and internal threats. To date, there have been no attempted assaults with the intent to commit radiological sabotage and, although there has been no PRA work done in the area of safeguards, it is assumed that there exists a small probability of an attempt to commit radiological sabotage. Although radiological sabotage is assumed to be a small probability, it is also assumed to be risk significant since a successful sabotage attempt could result in initiating an event with the potential for disabling of the safety systems necessary to mitigate the consequences of the event with substantial consequence to public health and safety. An effective security program decreases the risk to public health and safety associated with an attempt to commit radiological sabotage.

Statement of Objective

To protect against the design basis threat of radiological sabotage

Desired Result/Performance Expectation

Provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety. The physical protection program shall be designed to protect against the design basis threat of radiological sabotage.

I. <u>Physical Security Key Attributes</u>

The attributes that provide protection against the threat of radiological sabotage are shown in Figure 1. Those attributes provide the licensees the ability to provide defense in depth against both an external and an internal threat. Acceptable performance in the areas to be measures in the key attributes will provide assurance of the licensees' ability to protect against the threat of radiological sabotage.

Within each key attribute of the physical security cornerstone, specific areas for measurement either by performance indicators or by inspection activities have been identified for assessment.

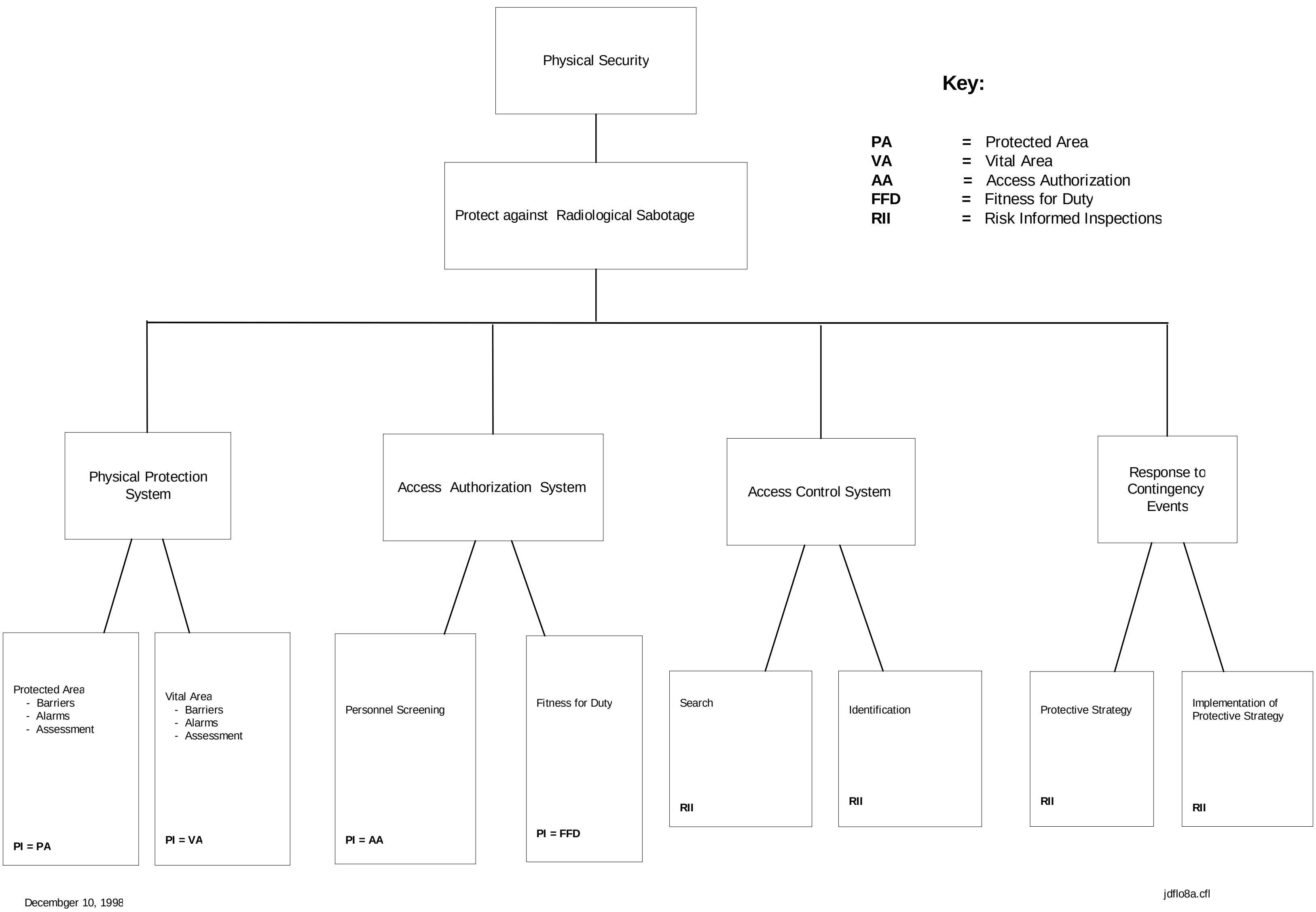


Figure 1

A. Physical Protection System

Within this key attribute, the areas to measure are barriers (protected and vital areas), Intrusion Detection System, and Alarm Assessment System. Operability of this system is necessary to detect and assess safeguards events and to provide the first line of defense in the defense-in-depth concept for protection against radiological sabotage. In the event of a malevolent act, the intrusion detection system identifies the existence of the threat, the barriers provide a delay to the person(s) posing the threat and the alarm assessment system is used to determine the scope of the threat. Data for the Physical Protection System are used to evaluate the scope of, and to initiate a response to, the threat.

Within this cornerstone attribute, PIs will be used to monitor the capability and availability of these systems to perform their intended function.

B. Access Authorization System

Within this key attribute, the areas to measure are the Personnel Screening process, the Fitness-for-Duty (FFD) program, and the Behavior Observation program. The personnel screening process is the process used to verify the trustworthiness of personnel prior to granting unescorted access to the protected area. The process includes psychological testing, a criminal history check, a background check, and reference checks with previous employers. The FFD program includes pre-employment, random, and for-cause testing for alcohol and illicit drugs. The Behavioral Observation program is conducted by supervisors and management personnel designed to detect behavior changes which, if left unattended, could lead to acts detrimental to the public health and safety. Within this area, data currently collected can be used for PIs to monitor the effectiveness of the implementation of the programs.

C. Access Control

Within this key attribute, the areas to measure are the effectiveness of the search function (personnel, package, and vehicle) and the Identification and Authorization process. The search function is to prevent the introduction of contraband (firearms, explosives, incendiary devices) that could be used to attempt to commit radiological sabotage. The search function also screens for prohibited articles, such as alcohol and illegal drugs. The Identification and Authorization process is to assure that, once personnel have been screened to verify their trustworthiness, those persons have a need for access and to confirm that only those persons who have been screened and have a need are granted access to the plant. During discussions between NEI, industry, and the NRC, it was concluded that meaningful tracking data on the performance of those processes was not practical since much of the performance is dependent on the quality of the implementation of the tasks. Assessment in this area will be through the baseline inspection process.

D. Response to Contingency Events

Within this key attribute, the areas to measure are the Protective Strategy and the implementation of the Protective Strategy. The protective strategy includes pre-identified target sets of vital safety equipment that must be protected to assure safe shutdown of the plant, a plan to get properly trained response personnel with the appropriate armament in place within pre-determined time lines in order to protect the plant against the design basis threat. The implementation of the protective strategy includes demonstrating that the strategy works and can successfully protect against the design basis threat through drills and exercises. Licensees conduct drills/exercises periodically but may not do so on a periodicity that provides a minimum number of valid data points during the course of the year that are considered necessary for a PI. Until there is a consistent approach that would provide these data points, assessment in this area will be through the baseline inspection process.

II. Performance Indicators

The following PIs are proposed to assess licensee performance in the Physical Protection and Access Authorization Systems. The PIs were selected based on their ability to provide objective measures of performance.

A. Physical Protection System

The performance for this system will be measured by the percent of the time all components (barriers, alarms and assessment aids) in the systems are available and capable of performing their intended function. When systems are not available and capable of performing their intended function, compensatory measures must be implemented. Compensatory measures are considered acceptable pending equipment being returned to service, but historically have been found to degrade over time. The degradation of compensatory measures over time, along with the additional costs associated with implementation of compensatory measures provides the incentive for timely maintenance/I&C support to return equipment to service. The percent of time equipment is available and capable of performing its intended function will provide data on the effectiveness of the maintenance process and also provide a method of monitoring equipment degradation as a result of ageing that could adversely impact on reliability. The reporting of equipment percent availability will be accompanied by the reporting of compensatory hours for equipment out of service due to equipment failure, the compensatory hours expended for equipment out of service due to extreme environmental conditions (severe storms, heavy fog, heavy snowfall, sun glare that renders the assessment system temporally inoperative, etc.) and for planned maintenance and modifications. The extreme environmental and planned maintenance and modifications compensatory hours will not be considered as equipment unavailability as part of the PI but are part of the total compensatory hours and will provide information on events that are contributing to equipment unavailability.

The tracking of equipment availability will provide an indication of the effectiveness of maintenance of the systems. Compensatory hours expended are currently tracked by the licensees, however they are not in all cases sorted by the categories proposed to be reported. Reporting this data should result in minimal additional burden. The data in this area will be reported as two PIs, one as the percent availability for the protected area system and one for the vital area system.

The thresholds for an acceptable percent availability for the systems are have been developed based on the professional judgment of the NRC, NEI, and industry peer working group on what is the appropriate level of performance. Industry historical data was collected in an attempt to benchmark the thresholds. However, because of the lack of consistency in past data collection and categorization processes obtaining a valid basis for the thresholds was not possible. The collegial decision on the thresholds by the industry group will be subject to review after data has been gathered for a period of time and some history is established. The thresholds that are proposed are 95-100% availability will be in the green band, 85-94% will be in the white band, and below 85% will be in the yellow band.

The thresholds should be reviewed after a 2 year period to evaluate their validity and make adjustments if necessary.

B. Access Authorization System

The performance indicator for this system will be the number of reportable events that reflect program degradations. This data is currently available and there are regulatory requirements to report significant events in the areas of Personnel Screening and FFD. The Behavior Observation significant events are captured in the FFD reporting requirements.

The thresholds for determining acceptable implementation of these programs were developed based on the professional judgement of the NRC, NEI, and industry peer working group. An attempt was made to benchmark the thresholds to validate the collegial decisions of the peer group. The bench marking data generally confirmed the perception that overall these programs were working as intended and did identify several programs with known weaknesses. However, because of a lack of consistency in the criteria used for reporting the data, complete confidence in the bench marking process was not possible. Standardization of the data reporting will be addressed during the V&V process. There will be 2 PIs for this area, one for access control and one for FFD. The thresholds that are proposed are: 0-2 reportable events per year in either area will put the area in the green band, 3-5 reportable events per year will put the area in the white band, and 6 or more per year will be in the yellow band. The thresholds will should be reviewed after a 2 year period to evaluate their validity and to make adjustments if necessary.

III. Inspection Areas

A. Physical Protection Areas

This area will be assessed by a PI after the initial V & V inspection is done to review the testing requirements for each system to assure performance standards and testing periodicity are appropriate to provide valid data for the PI.

B. **Access Authorization System**

This area will be assessed by a PI after the initial V & V inspection is done to confirm implementation is acceptable and that reporting thresholds for significant events meet regulatory expectations. The initial V&V inspection will serve to ensure valid data is used for the PIs.

C. Access Control

The areas of Search and Identification and Authorization will be inspected as part of the baseline inspection program. The inspection will consist of procedure reviews, self assessment reviews, and observation by the inspector. These are areas where the effectiveness of performing the task determines the effectiveness of the processes and also areas where the tasks are performed by numerous personnel in the security organization. Failure to properly perform the tasks could result in the introduction of contraband or unauthorized personnel into the protected area.

D. Response to Contingency Events

The areas to measure of Protective Strategy and Implementation of Protective Strategy will be inspected as part of the baseline inspection process. The inspection will consist of review of training and qualification records, the protective strategy, drill and exercise scenarios, drill critiques, and the results of a requested demonstration of the ability to defend against the design basis threat in order to prevent an act of radiological sabotage. This is the last line of defense in the physical security defense in depth process.

Table 1
Physical Security Key Attributes and Means to Measure

Key Attribute	Areas to Measure	Means to Measure	Comments
Physical Protection System	Barriers Intrusion Detection Alarm Assessment	PI V & V	Data on system availability and capability to detect and assess safeguards events
Access Control	Search Identification and Authorization	Inspection	The inspection will consist of procedure reviews, self assessment reviews, and observation by the inspector. Tasks in this area are performed by numerous personnel and the effectiveness of the process is dependent on the proper performance of the task to prevent the introduction of contraband or unauthorized persons into the protected area.
Access Authorization	Personnel Screening Fitness-for-Duty Behavior Observation	PI V & V	Data on process implementation is available to assess the performance in this area.
Response to contingency events	Protective Strategy Implementation of Protective Strategy	Inspection	The inspection will consist of review of training and qualification records, the protective strategy, drill and exercise scenarios, drill critiques, and the results of a requested demonstration of the ability to defend against the design basis threat in order to prevent an act of radiological sabotage. This is the last line of defense in the security defense in depth process.

Table 2
Physical Security Performance Indicators

PI	Measurement Area	Definition	Thresholds
1--Availability of protected area systems to perform their intended functions 2--Availability of vital area systems to perform their intended functions	Physical Protection Area • Barriers • Intrusion Detection • Alarm Assessment	Each PI is a percent of time the systems are available and capable of performing their intended function to detect and assess safeguards events	95-100% availability-green band 85-94% availability-white band 84% or less availability-yellow band
3--Acceptable implementation of the access authorization programs 4--acceptable implementation of the FFD & behavior observation programs	Access Authorization System • Personnel Screening • Fitness-for-Duty • Behavior Observation	Each PI is the number of reportable events that reflect problems in the implementation of the programs. These processes should be able to verify persons granted unescorted access to the protected area are trustworthy and reliable and not under the influence of any substance that adversely affects their ability to safely and completely perform their duties, and to provide reasonable measures for the early detection of any change in trustworthiness and reliability.	0-2 reportable events-green band 3-5 reportable events-white band 6 or more reportable events--yellow band

Appendix H

Supporting Analysis for Performance Thresholds for the Initiating Event and Mitigating Systems Cornerstone PIs

H.1 Introduction

The purpose of this appendix is to provide the results of analyses performed in support of the establishment of risk-informed performance thresholds. Section H.2 defines the scope of the analyses and PIs addressed. Section H.3 describes the analysis approach and the process by which the thresholds were established. The results are provided in Section H.4, and work still to be performed is discussed in Section H.5.

H.2 Scope

The analyses described in this appendix were performed to support the establishment of thresholds for those PIs for which PRA models could be used to provide a risk perspective, and for which industry-wide data were available. As will be discussed in Section H.3, in establishing the thresholds an important input was information on the range of values exhibited by the PIs across the plants. Not all the data required to achieve this was readily available and therefore thresholds were not developed for all the PIs that could, in principle, be addressed using PRA models. Data was provided by NEI on unplanned scrams and on the SSPIs, with the following exceptions: a) unavailability SSPI data was not provided for the PWR decay heat removal system, and while the raw data is available to the NRC, it was not possible to analyze it in the time available to complete this document, and b) data for the SSPI reliability indicators were not provided. Data on the occurrences of risk significant scrams were obtained from the draft report INEEL/EXT-98-00401, Rates of Initiating Events at U.S. Commercial Nuclear Power Plants, 1987 through 1995, April 1998. Thus the PIs for which analyses are reported in this appendix are:

- Initiating Event PIs
  - Reactor Trips
  - Risk Significant Scrams

- Mitigating Systems PIs - SSPI availability indicators for:

  - PWRs
    - Emergency Diesel Generators,
    - Auxiliary Feedwater System (AFW),

- High Pressure Injection System (HPSI),
- Decay Heat Removal System (RHR)*.

- BWRs
  - Emergency Diesel Generators,
  - High Pressure Coolant Injection System (HPCI, HPCS),
  - High Pressure Decay Heat Removal System (RCIC, IC),
  - Decay Heat Removal System (RHR).

* Because of the unavailability of data, for the PWR RHR system, the thresholds were established indirectly, by making use of those established for the BWR RHR system.

H.3 Approach to Establishing Thresholds

H.3.1 Overview

Three thresholds were established in accordance with the Figure 2 of the main report. The green-white threshold corresponds to declining performance, the white-yellow threshold to substantially declining performance, and the yellow-red threshold to unacceptable performance. When establishing the thresholds it was taken as guiding principles that they should not result in a large number of false positives (resource concern), and that thresholds should be set to capture meaningful changes.

PRA models were used to provide a risk-perspective on the thresholds. This was done by performing sensitivity studies to investigate how the core damage frequency (CDF) of the plants varies as the values of the PIs change. The analyses were performed by NRC staff or their contractors with the SAPHIRE code, using seven NRC-developed simplified models (SPAR models) and six licensee PRA models that were available at the INEEL. In addition, results from twelve licensee PRA models were provided by NEI. While, for most cases, the PRA results were able to provide information relevant to establishing the white-yellow and yellow-red thresholds, in some cases, the CDF results are insensitive to large changes in the parameters corresponding to the PIs. For these cases, an alternate approach to choosing thresholds was required.

H.3.2 Technical Issues

There are some technical issues related to the nature of PIs and PRAs that affect the way the PI data and the PRA results are used:

- The nature of the PIs for the initiating event and mitigating systems cornerstones is such that they are based on either the number of events, or the magnitude of events, or both, and therefore, each PI at a particular plant is subject to fluctuations with time.

- PRA models evaluate risk as a time-averaged quantity, based on the mean value of the parameters associated with the PIs. The statistical fluctuations discussed above are not accounted for even in an uncertainty analysis, which typically would address an epistemic (state of knowledge) uncertainty in the mean value, rather than the aleatory (statistical) variation.

- SSPIs are calculated using both unavailability associated with the planned removal from service for testing, preventive and correective maintenance and the estimated unavailability due to failure that occurred at some unknown time prior to being revealed. These SSPI values are not directly comparable to the parameters used for PRA basic events which model the planned unavailability and the failure contributions separately. In addition, the failure contributions are accounted for differently, using simple binomial or standby failure rate models for failures on demand and constant failure rate models for failures to run. The impact of this on the use of the PRA results is discussed in Section H.4.2.

- Thus there are two sources of variation to take into account when creating a decision model based on PRA input; the change in the (time-averaged) quantity provided by the PRA, and the statistical fluctuations. Therefore, to establish meaningful thresholds for the proposed indicators what is needed is:

  - a characterization of the range of values of the PI that denote acceptable performance, taking into account how those values fluctuate over different reporting periods.

  - the establishment of a relationship between the PIs and PRA model parameters that allows an assessment of how big a change in PI values is required to result in a risk-significant increase in CDF.

  For the PI to be a sensitive indicator of change, the change in the PI corresponding to a risk significant threshold has to be greater than the expected statistical fluctuations.

H.3.3 Approach

The following approach has been adopted.

The green-white threshold

To determine the green-white threshold, it is necessary to define what is acceptable performance. The green-white threshold for the PI was chosen to be commensurate with a generically achievable level of performance and takes into consideration the statistical variability arising from the random nature of the contributing events as seen across the entire population of plants. Data for the unavailability PIs was provided by NEI for all except the RHR system for PWRs. For the purpose of establishing the green-white threshold, histograms were provided of the maximum value recorded for each PI for all the plants (Figures H.1 through H.6). The threshold was determined by the simple approach of choosing a value to no more than two significant figures that is such that about 95% of the plants have observed data values that would be in the green zone, and is therefore established on a generic basis. This method depends only on the number of plants with less than acceptable performance, but not on determining by how much their performance exceeds the norm. Alternative approaches, such as using the mean plus two standard deviations of the PI values to set the threshold puts more weight on the actual values of the PIs, and could be biased by the poor performers in a non-conservative direction. This threshold value may be higher or lower than the value of the corresponding parameter used in licensee’s PRAs. That the threshold is reasonable from a risk standpoint was demonstrated by the fact that use of the threshold in the sample of PRA models used for the sensitivity studies would have resulted in an increase in CDF of less than 1E-05/reactor year.

The white-yellow threshold

There is no clear regulatory definition of unacceptable risk in numerical terms that can be used to calibrate declining or unacceptable performance. However, in RG 1.174, the NRC has established acceptance guidelines for allowing changes to the licensing basis that relate to changes in CDF and LERF. Specifically, for CDF, an increase in the range of 1E-06 to 1E-05/reactor year would be acceptable, under certain conditions and with staff review and approval, while changes resulting in an increase greater than 1E-05/reactor year would not be acceptable. While these acceptance guidelines are intended for permanent changes to the licensing basis, it would be consistent to also apply these to changes resulting from operating practices, using the argument that if the degradation in performance were uncorrected, it would lead to a permanent increase in CDF. Furthermore, a change in CDF of 1E-05/reactor year is used in the staff’s regulatory analyses as one element in determining the requirement for a backfit. Thus, it was decided that the white-yellow threshold should be determined on the basis of sensitivity analyses to identify that mean value of the PRA parameter associated with the PI that would increase CDF by an amount that corresponds to a substantially declining performance, which has been chosen as 1E-05/reactor year. For the PI to be a meaningful indicator, this increase must be significant compared with the expected statistical variation captured by the setting of the green-white threshold. In comparison with the way the green-white threshold is determined, this approach is somewhat conservative in that it does not increase the value to compensate for the expected statistical variation. However, since this is only an indicator of performance rather than a criterion for regulatory action, this is considered appropriate.

The yellow-red threshold

A truly unacceptable performance would likely correspond to a change in CDF well in excess of 1E-05/reactor year, and is chosen as corresponding to a change in CDF of 1E-04/reactor year. The yellow-red thresholds were determined by identifying the PI values that would correspond to increases in CDF of 1E-04.

H.3.4 General Discussion of Approach

The results of the sensitivity analyses indicated variability that could be associated to some extent with design differences although there is also variability due to PRA modeling differences. Therefore, while it is suggested that different values may be appropriate for the thresholds for some SSPIs depending on the degree of redundancy of the associated system, or depending on the plant or system design features, the sample of studies performed in support of this activity is too small to be definitive, and further work is needed as described in Section H.5.

While no data were provided with which to investigate the reliability green-white thresholds, sensitivity studies were performed for the reliability parameters of PRAs. As discussed in Section H.4.2, because of the way in which the unavailability SSPIs are evaluated, these studies were used in the determination of thresholds for the unavailability SSPIs for redundant systems.

Because the models used for the sensitivity studies did not provide an easy way to calculate LERF, they do not provide a complete risk perspective. However, the PIs addressed here are associated with the initiating event and mitigating system cornerstones only. The containment issues are addressed in the barrier cornerstone.

In addition to the sensitivity studies performed to establish the thresholds, a limited number of studies were performed to investigate the risk impact of more than one of the PIs increasing simultaneously, and are discussed in section H.4.4.

H.4 Results

H.4.1 Initiating Events

Two PIs are proposed; the number of unplanned scrams in 7000 critical hours; and the number of risk-significant scrams in a three year period.

H.4.1.1 Number of Unplanned Scrams

There is a direct relationship between this PI and a parameter in the PRA models, namely the frequency of initiating events. However, in performing the sensitivity studies, a simple scaling of all initiating event frequencies by the same factor, would result in a proportionate increase in CDF. Bearing in mind that the purpose of this indicator is to determine when it is appropriate for NRC to initiate a response, and that initiating events are not all equal in their risk significance, it was considered more meaningful to perform sensitivity studies by increasing the frequencies only of those initiating events that are expected to occur. Therefore the frequencies of those rare, but potentially risk significant initiating events such as LOCAs, SGTR, LOSP, and failure of a support system were not increased when performing the sensitivity studies. If any of these potentially risk significant scrams were to occur, it is highly likely that a reactive inspection would be initiated.

The data obtained from NEI (Figure H.1) for the number of unplanned scrams indicates that, for the vast majority of plants, the frequency of unplanned scrams has consistently been less than 3 per year, for the last three years. Furthermore, the data in the draft AEOD study on initiating events (INEEL/EXT-98-00401, April 1998) indicates that the averge number of scrams is 2.1/reactor year. These two pieces of evidence argue for setting the green-white threshold at 3. (Even though there may be minor differences in the evaluation of the denominators of these estimates, this will not make a significant difference to the conclusions drawn here.)

Since the average value for scram frequency used in the IPEs is 7.4/year based on the AEOD report, a generic value of 3 for the threshold will not constitute a concern about the level of risk.

The results in Table 1 indicate that the number of scrams that would lead to a change in CDF greater than 1E-05/reactor year is somewhere in the range of 5 to greater than 10. The numbers in parentheses for the Palo Verde, Brunswick, and Comanche Peak IPE models, reflect the results of using a smaller set of initiating events that correspond to those relatively uncomplicated scrams, such as reactor trips, that are expected to occur more frequently. Based on the results in Table 1, the white-yellow threshold is proposed to be set at 6, with the caveat that, if a scram results from an event that is caused by a loss of a critical function (heat removal, pressure boundary) or loss of a support system, it will be subject to a reactive inspection. One of the NEI plant studies resulted in a frequency of only 4.5, but these studies were performed by increasing the frequency of all scrams (including LOSP) by the same factor, thus distorting the picture with respect to the more common reactor trips.

The studies indicate that the yellow-red threshold is at such a high value that it is realistically unachievable.

H.4.1.2 Number of Risk-Significant Scrams

Data from the INEEL draft report INEEL/EXT-98-00401 on the number of scrams which involved more than simply a reactor trip suggest that an appropriate value for the green-white threshold is 4 events in three years, or 1.33 per reactor year.

The sensitivity studies were performed by increasing the frequency of a selected number of the initiating event used in the PRA models, namely those representing loss of the power conversion system. The Loss of Offsite Power and losses of support systems were not included since they have a disproportionate impact on CDF and furthermore are relatively more rare, and would in any case initiate a more significant regulatory response. The results of these studies are given in Table 2.

The results for Palo Verde are considerably lower than those for the other plants, and this is largely due to the design of the plant, which does not allow feed and bleed as an alternate method of decay heat removal, relying entirely on the auxiliary feedwater systems. Based on these results, the white-yellow threshold is set at 10 per three years for all plants except those for which feed and bleed is not an option. These plants will be treated in a design specific way. The yellow-red threshold is again significantly higher and is realistically unachievable.

H.4.2 Unavailability

The SSPI unavailability indicator does not have a one-to-one correspondence with a parameter in the PRAs. In PRAs, the unavailability parameter typically only represents the ratio of the time the train was out of service (tagged out) to the time required to be available. The SSPI includes a contribution from the so-called fault exposure time, which is an estimate of the time the train was unavailable due to a failure before that failure became revealed. Thus it contains some of he impact of the failures which would be included in the failure to start and run events in the PRA. Thus in performing the sensitivity analyses for redundant systems the correct way is intermediate between the two approaches below:

a) increase the unavailability basic event. This is somewhat of an underestimate because typically multiple unavailability cutsets are deleted as being disallowed by tech specs.

b) increase the failure to start and run contribution (including the CCF term if necessary). This is conservative because it is expected that a significant contribution to the SSPI is from the out-of-service unavailability that would not occur simultaneously in multiple trains as it is limited by technical specifications.

For the single train systems such as HPCI, HPCS, and RCIC, the two are equivalent in their impact on CDF.

H.4.2.1 BWR High Pressure Injection Systems (HPCI, HPCS) and High Pressure Core Cooling System (RCIC)

The SSPI data provided by NEI suggests a suitable value for the green-white boundary is .04 (Figure H.2). This is on the same order as the value used for the total unavailability on demand of a HPCI or RCIC train used in PRAs (failure to start and run and unavailability due to maintenance). It is also on the same order as typical values used for the HPCS system including

Table 1
Sensitivity to Number of Scrams

Plant	Model type	Base Case frequency	Frequency giving increase in CDF of >1E-05/yr	Frequency giving increase in CDF of >1E-04/yr
Perry	SPAR	2.63	9	>50
Brunswick	SPAR	2.8	7.3	>50
Crystal River	SPAR	2.28	>10	>100
SONGS	SPAR	2.49	6.4	~50
Kewaunee	SPAR	2.25	>10	~35
North Anna	SPAR	2.52	>10	~45
Seabrook	SPAR	2.26	>10	>100
Surry	IPE	2.75	>10	>100
River Bend	IPE	4.42	>10	>100
Crystal River	IPE	-	-	-
Palo Verde[1]	IPE	3.3 (2.85)	5 (6)	<30
Brunswick[1]	IPE	2.4 (2.27)	8.6 (11.35)	>100
Comanche Peak[1]	IPE	4.3 (2.9)	5-6 (8.2)	
Plant 1 (PWR)	NEI	3	10	*
Plant 2 (PWR)	NEI	1.4	4.5	*
Plant 3 (PWR)	NEI	5	34	*
Plant 4 (PWR)	NEI	7	27	*
Plant 5 (PWR)	NEI	2.75	70	*
Plant 6 (PWR)	NEI	1.95	10.3	*
Plant 7 (PWR)	NEI	1.1	9.1	*

Plant 8 (PWR)	NEI	1.04	4.66	*
Plant 9 (BWR)	NEI	4	10	*
Plant 10 (BWR)	NEI	6.23	18.8	*
Plant 11 (BWR)	NEI	1.5	22	*
Plant 12 (BWR)	NEI	.71	9.12	*

* NEI plant studies include all trips (not LOCAs). Results were not provided for changes in CDF of 1E-04.

[1] The numbers in parenthesis result from using the subset of uncomplicated scrams (see text).

Table 2
Results of Sensitivity Studies for Risk-Significant Scrams

Plant	Base Case Frequency	Frequency giving an increase in CDF >1E-05/year	Frequency giving an increase in CDF >1E-04/year
Brunswick	.77	5.2	46
River Bend	2.76	~70	>100
Palo Verde	.46	1.15	7
Surry	.15	3.6	34
Comanche Peak	1.41	3.5	21

Note: The IPE model for Crystal River had no general transient category contribution.

the HPCS diesel generator system.

The sensitivity studies result in a range of values for the white-yellow boundary, with the lowest value being at about the .04 value (see Table 3). This value, for the Perry plant, was, however, specifically for the HPCS pump excluding the diesel generator. Since the HPCS pumps typically demonstrate a lower unavailability than do the steam-driven HPCI pumps, this suggests that the indicators should be treated separately. The results show that, for all the other plants, there is significant margin between the value associated with the white-yellow threshold and either the PRA values assumed or the white-green threshold value, and on the basis of these results, a white-yellow threshold of .12 is proposed, with a yellow-red threshold of .5 for HPCI. For HPCS, when the diesel generator is excluded, the threshold should be lower. The white-yellow threshold is proposed as .04, with the yellow-red threshold set at .2.

The data provided by NEI averaged the data between the HPCI/HPCS and RCIC/IC systems. Therefore, the thresholds for RCIC are the same as those for HPCI.

H.4.2.2 Diesel Generator Unavailability:

The SSPI data provided by NEI suggests establishing the green-white boundary at .025 (Figure H.3).

The results of the sensitivity studies performed in support of establishing thresholds are given in Table 4. As discussed above, there is no direct correspondence between the SSPI indicators and the parameters used in PRA models. Therefore, to explore the risk implications of variation in the indicator values two sets of sensitivity studies were performed using the IPE models: 1) the sensitivity studies designated as "unavailability" were performed by changing the basic events in the PRA models that represented unavailability due to test and maintenance. The PRA models delete cutsets that represent two or more trains out of service at the same time. 2) the sensitivity studies designated as "unreliability" were performed by changing basic events representing failure to start, failure to run, and common cause failure by the same factor. These studies indicate that for the unavailability parameter, the four train plant (Brunswick) and three train (Surry) allow a larger increase over the base case than the two train plants. For the unreliability studies, this difference is not as marked, although for the four diesel plants the factor increase over the base case is generally higher than for the two diesel plants (3 and higher versus 2). For the four diesel plants the common cause failure models play a more significant role and the increase in CDF is more linear with the increase factor than for the two diesel plants.

The SPAR models are not well suited for performing these studies as there is only one term representing the sum of failure to start and run and unavailability. Furthermore, the modeling of common cause failures is fairly conservative compared to licensee-generated PRA models. The NEI studies were performed by increasing the values for one train at a time and therefore, with the exception of plant 1, the results for unavailability and unreliability were comparable. The conclusions here are based on the studies performed with the IPE models. Previous studies performed in NUREG 1032 demonstrated that the impact of increasing the unreliability of a single train has a bigger effect on two train versus three or four train systems, even taking into account the effect of common cause failures. Basically the reason is that for the two train systems the contributors to system failure probability from the independent failures, (p^2) are comparable to those from the CCF term (βp), whereas for the higher redundant systems the dominant term is the CCF term ($\gamma\beta\delta p$ for a four train system using a multiple Greek letter parameterization of CCF probability), and the p^4 terms are much less significant. The impact increases with unreliability. Because of this, and guided by the sensitivity studies performed, it is recognized that there probably ought to be a different threshold for the plants with higher redundancy. The sensitivity studies performed suggest that, for the interim, for two train plants, a white-yellow threshold of .05 and a yellow-red threshold of .1 be used, and, for plants with three or more trains, the white-yellow threshold is set at .1, with the yellow-red at .2. Because this is based on a sample of only one for the four train plants, this should be confirmed by performing sensitivity studies for more plants.

If a reliability PI is developed, it would then be appropriate to redefine the unavailability PI to reflect the "tagged" out-of-service contribution, and this would allow a significant distinction between the new unavailability indicators for higher redundancy plants.

Table 3

Sensitivity of CDF to BWR HPI unavailability

Plant	Model type	System	Base Case Parameter Value	parameter value for ΔCDF >1E-05	Ratio Column 5:threshold value	Ratio Column 5 to column 4	parameter value for ΔCDF >1E-04	Ratio Column 8 to column 4
Perry	SPAR	HPCS	3.7E-03	3.9-02	~1	>10	>.2	5
		RCIC	3.2E-02	.3	7.5	~10	1	33
Brunswick	IPE	HPCI	.0126	>.13	3.25	>10	>.5	40
		RCIC	.0106	.5	12.5	50	>.5	>50
River Bend	IPE	HPCS[1]	.04	>.4	10	>10	>.75	>15

		RCIC	.058	~.4	10	7	~1	
Plant 9	NEI	HPCI	.018	.13	3.25		**	
		RCIC	.022	.23	5.75		**	
Plant 10	NEI	HPCI	.026	.48	12		**	
		RCIC	.025	.58	14.5		**	
Plant 11	NEI	HPCI	.0126	.48	12	38	**	
		RCIC	.01	.43	11	43	**	
Plant 12	NEI	HPCI	.0075	.5	12.5	66	**	
		RCIC	.0045	.48	12	>100	**	

Notes: ** For NEI plants, only the change in unavailability giving ΔCDF = 1E-05 was provided

[1] The results for this sensitivity study include the HPCS diesel generator, and from the point of view of unreliability correspond more closely to the HPCI system.

Table 4
Sensitivity Studies for Diesel Generators

Plant	Model type	Parameter type	Base Case Parameter value	Parameter value for ΔCDF>1 E-05	Ratio column 5:threshold value	Ratio column 5:column 4	Parameter value for ΔCDF>1E-04	Ratio column 8:column 4
Perry	SPAR	Unavailability	.038	~.12	4.8	-	1	
		Reliability	.038	.08	3.2	2	.19	5
Brunswick	SPAR	Unavailability	.037	~.14	5.8	-	1	
		Unreliability	.037	~.05	2	1.35	~.1	
Brunswick	IPE	Unavailability	.038	.38	15.3	10	1	
		Unreliability	.04	.12	5	3	>.25	
River Bend	IPE	Unavailability	.028	~.06	2.4	2.15	1	
		Unreliability	.12	~.2	8	1.7	.7	6
Surry	IPE	Unavailability	.01	~.06	2.4	6	.5	
		Unreliability	.024	.048	2	2	~.1	5
Palo Verde	IPE	Unavailability	.0105	.04	1.6	4	.2	
		Unreliability	.017	.043	1.7	~2.5	.17	10
Comanche Peak	IPE	Unavailability	.015	~.045	1.8	3	~.3	~20
		Unreliability	.096	.18	7.2	2	~.5	5
Crystal River	IPE	Unavailability	.0063	.16	6.4	~10	>.5	
		Unreliability	.0214	.06	2.4	3	.6	10
Plant 1	NEI	Unavailability	.019	.022			**	

Plant 2	NEI	Unavailability	.012	.103			**	
Plant 3	NEI	Unavailability	.018	.67			**	
Plant 4	NEI	Unavailability	.014	.26			**	
Plant 9	NEI	Unavailability	.006	1			**	
Plant 10	NEI	Unavailability	.03	1			**	

H.4.2.3 BWR Residual Heat Removal System:

The SSPI data provided by NEI suggest a green-white threshold of .015 (Figure H.4). This, however, is higher than unavailabilities typically used in PRAs, which are in the range .003 - .01. The PRA evaluations from both IPE and SPAR models showed that the CDF was insensitive to the unavailability values, largely because common cause effects and the assessment of the failure of the operators to initiate RHR dominate. While the SPAR models showed some sensitivity to the reliability values, the IPE models did not. PRA models are not likely to give much insight into the setting of the thresholds, because of the way the function is typically modeled as being backed up be containment venting, and because of the long time available for taking action. Therefore, alternate approaches are required to set the thresholds. One approach is to set the white-yellow threshold using the AOTs. The threshold is therefore set at .05, which corresponds to about two occurrences of unavailability corresponding to an AOT of 7 days in any one year.

H.4.2.4 PWR Auxiliary Feedwater Systems:

NEI provided data that represented an average train unavailability, which in many cases included a mixture of motor driven and turbine driven pump trains. Data collected on turbine driven and motor driven pumps indicate that both the reliability and unavailability are different by as much as an order of magnitude. Furthermore, PRA results have shown that the turbine driven train is important to cope with medium term station blackout scenarios. Because of this, it is proposed that the indicators should be provided separately for turbine driven and motor driven trains.

The current averaged data suggests a green-white threshold of .02 (Figure H.5), although it should perhaps be less for the motor driven trains. The value of .02 is on the order of the failure probabilities (unavailability + unreliability)used for motor driven pump trains in PRA models whereas experience data suggests a lower value (see NUREG/CR 5500, Vol. 1). The failure probabilities used in IPEs for turbine driven trains are about a factor of 5 higher. The experience captured in NUREG/CR 5500 shows a bigger difference with the failure probability of the motor driven trains being lower and that of the turbine driven trains being higher than assumed in PRAs. Therefore, perhaps the threshold for the turbine driven train on its own should be higher.

Table 5 summarizes the results of sensitivity studies performed on AFW model parameters. Since, for all the models used, there is only one turbine driven train, only the unavailability sensitivity is reported. For the motor driven trains, when there is more than one train, both unreliability and unavailability results are presented. The data are difficult to interpret, partly because of differences in the way the systems are modeled, and the assumptions made about the turbine driven train running failure rates. However, the table indicates that there is a margin between the values used in PRAs and those values that would give an increase of 1E-05 to the CDF of at least a factor of three. Thus, as interim values, the white-yellow threshold is set at .06. The yellow-red threshold could be considerably higher, but for now it is proposed to be set at .12. Note that, because of its design features that do not allow feed and bleed as a means of decay heat removal, the threshold for the Palo Verde plant must be established differently, and considerably lower.

H.4.2.5 PWR High Pressure Injection Systems:

The data supplied by NEI suggest a green-white threshold of .015 (Figure H.6). Very few of the sensitivity studies showed any significant impact on CDF, and therefore, the PRA models are not useful for setting the thresholds. The reason varies, and could be a function of the modeling, or of the fact that the HPSI pumps are in some plants backed up by charging pumps, or in other plants they double up as charging pumps, which eliminates the failure to start as a failure mode for one of the redundancies.

In any case, the same approach is taken to set the white-yellow threshold as in the case of the BWR RHR system.

H.4.2.6 PWR RHR Unavailability

In the absence of data on the SSPI, the green-white threshold was chosen to be the same as that for the BWR RHR system, i.e., .015.

The results of the sensitivity studies performed using the PRA unavailability term showed very little sensitivity of CDF to significant changes in the parameter. This was also the case for the sensitivity studies performed using the reliability parameters for those sensitivity studies performed by NEI. This was to be expected as these studies are performed changing the parameters for only one train. Sensitivity studies performed using the licensees' PRA/IPE models again showed that the unreliability parameters could be increased by a significant amount before changing CDF significantly. The SPAR models showed a greater effect. This appears to be due to the higher CCF probabilities used in those models compared to those used in the PRA models. Lower values than those used in the SPAR models are supported by the AEOD CCF data base. There is probably a case to be made for differentiating between those plants that require RHR pumps for high pressure recirculation and those that do not. However, it has to be borne in mind that in many cases PRAs model the success path for small LOCAs as sump recirculation rather than initiation of RHR. The impact of this needs to be explored before setting thresholds.

For the interim, it is suggested that the thresholds be the same as those for the BWR RHR system.

Table 5
AFW sensitivity studies

Plant	model type	Pump driver	Parameter	Base case	Value that gives ΔCDF = 1E-05	Value that gives ΔCDF = 1E-04
Crystal River	SPAR	Turbine		3.2E-02	.15	
		Motor		3.9E-03	no change	no change
San Onofre	SPAR	Turbine		3.7E-02	7E-02	~.1
		Motor	Unavailability	4.5E-03	.1	>.5
			Unreliability	*	x 1.5	
Kewaunee	SPAR	Turbine		3.2E-02	no change	no change
		Motor	Unavailability	3.9E-03	no change	no change
			Unreliability	*	> x10	
North Anna	SPAR	Turbine		3.5E-02	.4	?
		Motor	Unavailability	3.8E-03	no change	no change
			Unreliability	*	> x10	
Seabrook	SPAR	Turbine		3.3E-02	.15	1
		Motor	Unavailability	1.0E-2	>.5	1
			Unreliability	*	~ x4	
Comanche Peak	IPE	Turbine		2.7E-02	6E-02	1
		Motor	Unavailability	3.7E-03	.1	?
			Unreliability	2.5E-2	~7.5E-02	
Palo Verde[1]	IPE	Turbine		4E-03	4E-02	>.2
		Motor	Unavailability	7.5E-3	7E-2	

			Unreliability	2.5-03	7.5-03	>x10
Surry	IPE	Turbine		1.93E-02	1.9E-1	1
		Motor	Unavailability	2.7E-03	4E-02	>.2
			Unreliability	7.5E-03	>7.5E-02	
Crystal River	IPE	Turbine/ Motor	Unavailability/ Unreliability		No data	

Notes to table 5: for the SPAR models, the unreliability calculations were performed multiplying all trains by the same factor. The parameter values are the same as those given for the turbine driven and motor driven trains.

[1] Note that the unreliability required to produce a change in CDF of 1E-05 is considerably lower than that for the other PWRs, again because of the design features of the Palo Verde plant.

H.4.3 Summary

The results of the threshold evaluations are summarized in Table 6.

Table 6
Summary of Thresholds

Performance Indicator	Green-White threshold	White-Yellow threshold	Yellow-Red threshold
Number of Unplanned Scrams per 7000 critical hours	3	6	>25
Number of Risk-Significant Scrams per three year period	4	10	>20
SSPI BWR HPCI unavailability (also used for RCIC), or HPCS (diesel generator included)	.04	.12	.5
SSPI BWR HPCS unavailability (diesel generator excluded)	.015 (based on RHR/HPSI)	.04	.2
SSPI EDG unavailability	.025	.05 (2 EDGs) .1 (>2 EGDs)	.1 (2 EDGs) .2 (>2 EDGs)
SSPI BWR RHR unavailability/ SSPI PWR RHR unavailability	.015	.05	TBD
SSPI PWR AFW unavailability	.02	.06	.12
SSPI PWR HPSI unavailability	.015	.05	TBD

Note: Numbers in parentheses for EDG unavailability refer to plants with three or more diesel generators.

H.4.4 Sensitivity to Multiple Changes

Several sensitivity studies were performed to investigate the impact of an increase in more than one of the PIs simultaneously. The effect of two PIs being increased by the same factor is not surprisingly greater than either of the two taken individually. However, the result is not multiplicative but more nearly additive. However, the sensitivity analyses also demonstrated that the white-yellow threshold for all PIs was not determined by the same plant. Sensitivity analyses were performed for two of the plant models, Surry and Brunswick, to demonstrate the following: first, that setting all the PIs at the green-white threshold does not imply the risk will be increased significantly; and second, that setting all the PIs at the white-yellow thresholds can result in a significant increase in CDF. The results are tabulated in Table 7.

These results deserve some comment. The fact that the increase for Surry is not in excess of 1E-05 even when all the PIs are taken to their white-yellow threshold is a result of the fact that the Surry model was not limiting in determining any of the thresholds. However, as the Brunswick results indicate, the impact can be significant. This can be taken as a demonstration that two or more PIs in the white band should be treated as being more significant than only one being in the white band.

Table 7
Results of changing all PI parameters

Plant	Increase in CDF with all PIs at the green-white threshold	Increase in CDF with all PIs at the white-yellow threshold
Brunswick	8.56E-07	1.4E-05 (using 2 EDG threshold) 2.14E-05 (using the >2 EDG threshold)
Surry	2.51E-07	9.91E-06

H.5 Further Work

The current assessment of PI thresholds is based on a relatively small number of sensitivity studies, using PRA models of differing levels of detail. They show significant differences in results. It is desirable to broaden the range of sensitivity studies to include a greater number of plants. More effort is needed to understand these results, and to determine whether it is reasonable and meaningful to establish different thresholds for classes of plants that are identified by specific design differences such as the degree of redundancy, or even whether it would be more desirable to make them plant-specific. In addition, the reliability characteristics of turbine driven pumps appear to be sufficiently different from those of motor driven pumps that it makes sense to look at them separately. The following specific activities are proposed:

- (contingent) analysis of the results of sensitivity studies provided by NEI, performed in accordance with the approach used by NRC staff, and in particular to determine the feasibility of establishing more firmly different thresholds plants with different degrees of redundancy, and with different design features.

- separate analysis of data on turbine driven pump and motor driven pump trains (again contingent on receiving data from industry). In addition, for those plants that do not have feed and bleed capability, different thresholds need to be established.

- analysis of PWR DHR data to establish thresholds.

- analyses to support the establishment of thresholds for new PIs, or refinement of PIs.

Figure H.1

Figure H.2

Figure H.3

Figure H.4

Figure H.5

Figure H.6

APPENDIX I

BENCHMARKING THE PERFORMANCE INDICATORS

INTRODUCTION

Benchmarking is one method used for validating the PIs. It involves collecting and analyzing PI data for plants that NRC senior managers have previously identified as watch list plants, declining performers, and superior performers. The purpose of benchmarking is to determine if the PIs can (1) differentiate between plants perceived as superior, average, declining, and poor performers, and (2) identify declining performance in a timely manner so that increased regulatory attention can be applied before performance becomes unacceptable. Performance differences between plants would be reflected in the number and magnitude of PIs in the Regulatory Response (white) and Required Regulatory Response (yellow) bands for each plant. Timely response would be evidenced by declining trends prior to the occurrence of risk-significant events; the goal is that the PIs go through the white band before reaching the yellow band, with only a few instances of a PI going directly from green to yellow, and that PIs will very rarely go directly from green to red. The validity of this benchmarking process is based on the assumption that the Senior Management Meeting process identified the right plants at the right time.

INITIATING EVENTS, MITIGATING SYSTEMS, AND BARRIERS CORNERSTONES

Background

NEI performed a benchmarking analysis on a set of eight plants that they categorized as excellent, average, or declining performers, plus eight NRC watch list plants. These plants were identified by the letters A through P. The indicators they used are the ones originally proposed in their draft white paper, *A New Regulatory Oversight Process,* dated July 27, 1998, (RCS Activity, RCS Leakage, Containment Leakage, Unplanned Scrams, Safety System Actuations [SSAs], and Transients) except the Reliability and Availability of Risk-Significant systems, structures, and components and Shutdown Operating Margin. Since NEI did not have unavailability data at the time, they used Safety System Failures (SSFs) from the NRC PI program as a surrogate. They used monthly or quarterly data from July 1995 through June 1998 for RCS activity, RCS leakage, and containment leakage provided by the plants. NEI also used annual data from 1990 to 1997 on Scrams, Safety System Actuations, and Safety System Failures (SSFs) from AEOD annual reports, and data from 1990 to 1995 on Transients from an NUS database of licensee monthly reports. All of this data had been plotted by NEI and provided to the NRC. (NEI subsequently received and plotted SSPI data from the third quarter of 1995 through the second quarter of 1998.) NEI documented insights they had gleaned from their analysis of these data, including typical PI characteristics for each plant performance category which showed a correlation between the PIs and performance. These insights were obtained primarily from the SSF and Transients indicators. They concluded that the set of indicators provides an overall perspective of safety performance, and that the indicators do distinguish between levels of performance in enough of the indicators simultaneously to be a viable assessment tool.

Scope
The staff reviewed the NEI benchmarking analysis and performed its own independent analyses to answer the following questions:

3. Do the PIs as a set differentiate between superior, average, declining trend, and watch list plants as designated by the SMM process?
4. How effective are individual PIs at differentiating between plants with different levels of performance as designated by the SMM process?
5. Do the PIs demonstrate timely response (i.e., do not go directly from green to red)?
6. Do the PIs show declining trends for plants in SMM designated performance categories prior to SMM actions? If so, which ones are most effective ? If not, would they be expected to show a declining trend?
7. Do the PIs show declining trends prior to ASP events? If so, which ones are most effective?
8. How well does the set of PIs conform to those selected by Arthur Andersen for use in the trending methodology currently being used in the SMM process?
9. Do small decreases in the green-white thresholds capture more of the watch list and declining trend plants (sensitivity analyses)?

To perform its analyses, the staff selected a set of 17 plants, including three plants identified by the NRC senior management meeting as superior performers, four average plants, four plants that have received trending letters from the NRC, and six watch list plants (see Table 1).

While NEI used the PIs they originally proposed, the workshop and public meetings that have been held since then have resulted in agreement on the set of indicators described herein. There were a number of changes in both PI definitions and calculational methods. Also, NEI in their analysis used annual rather than quarterly values. In addition, there are some inconsistencies in the data from plant to plant and from year to year. However, the staff believes that these PIs are close enough to the proposed set of indicators to be useful for this benchmarking effort. Additional analysis and refinement of the PIs will be performed during the pilot program.

Table 1. NRC Benchmark Plants

Superior:	*Average:*
Callaway	Davis-Besse
Vogtle 1&2	Point Beach 1&2
	TMI 1
Trending:	*Watch List:*
Cooper	Crystal River 1
D.C. Cook 1&2	Indian Point 3
Hope Creek	Maine Yankee
	Millstone 1, 2, 3

Because barrier PI (RCS Activity, RCS Leakage, and Containment Leakage) data are not readily available to the staff (they are not required to be reported) and to expedite the process of data compilation, calculation, and plotting, NEI assisted the staff in collecting the data from publicly available sources for our analyses.

The staff received quarterly SSPI values from INPO for mid-1995 through mid-1998 that, since the SSPIs are averaged over a three year period, included data from 1992. The staff also obtained monthly or quarterly Barrier data from mid-1995 to mid-1998 and annual Scram, SSA, SSF, and Transient data from 1990 to 1997.

Although some data back to 1990 were available, because industry performance improved substantially between then and 1993, the staff found that the earlier data dominated the results and masked more recent plant performance. Therefore the staff used data from 1993 on. For its analyses, the staff selected plants whose SMM designations had changed between 1994 and 1998, to see if declining or improving trends could be identified before and/or after a plant's status changed. This included plants added to the list of superior performers (Callaway, Vogtle), plants that received trending letters (D.C. Cook) and then corrected the adverse trend (Cooper, Hope Creek), and plants added to (Crystal River, Maine Yankee, Millstone) or removed from (Indian Point 3) the watch list.

Method

The staff analyzed the data for the set of plants selected by NEI and the set selected by the staff, and developed Tables 2 and 3 to help find answers to the seven questions. Table 2 shows the lowest performance band and the highest value for each PI from 1993 on for which data were available. These values did not necessarily occur in the same year. Table 3 shows those plants that had more than one PI in the white or yellow bands and the year during which they occurred. The staff also looked at the minutes of the SMM discussions and talked to the appropriate program managers to understand why the selected plants were discussed, sent trending letters, or placed on the watch list. There were of course a number of reasons, some of which we would have expected the PIs to identify and others that would only be found by inspection. In Table 2, under the Method of Problem Identification, we have listed the method that we concluded was more likely to have identified the problems, although both PIs and inspection would have been involved. We also looked at each plant's PIs the year before the SMM action to see if they provided a leading indication. Table 2 was useful in analyzing the effectiveness of the PIs as a set and individually, in performing sensitivity analyses of the thresholds, and in identifying plants in the yellow band to assess the timeliness of the PIs.

Table 3 was useful in identifying degraded cornerstones or problems in more than one cornerstone simultaneously .

For plants that had an ASP event of $>10^{-5}$ conditional core damage probability (CCDP) between 1995 and 1997, the watch list plants, and the declining trend plants, the staff looked at the PIs in the year prior to the event or NRC action to see if the PIs provided a leading indication. For each of the plants that had more than one white and/or yellow PI, we noted the year of occurrence of each to look for a concurrent decline in more than one PI for a single cornerstone, or degradation in two or more cornerstones simultaneously. In addition, the staff reviewed the correlation analyses performed by Arthur Andersen in developing the trending methodology currently used in the SMM process to determine if the proposed PIs are among those that would be expected to identify SMM discussion plants.

Results

The staff's analyses resulted in the following findings with regard to the seven questions addressed in this study:

Table 2. Benchmarking Summary

Performance Category	Plant	RCS Activity[1]	RCS Leakage[1]	Contain-ment Leakage[1]	Scrams[2]	SSAs[2]	SSFs[3]	Transients >20%[4]	Safety System Performance Indicator (Unavailability)				Method of Problem Identification
									EAC[5]	HPI[6]	AFW[7]	RHR[8]	
Excellent (NEI)	A	G	G	G	G (1)	G (1)	G (1)	G(2)	G (0.011)	G (0.012)	G (0.007)	NA	NA
	B	G	G	G	G (1)	G (1)	G (3)	G (7)	G (0.007)	G (0.009)	G (0.018)	NA	NA
	C	G	G	G	G (2)	G (1)	G (2)	W (12)	G (0.007)	G (0.011)	G (0.013)	NA	NA
Superior (NRC)	Callaway	W (1Q)	G	G	G (3)	G (0)	G (3)	G (5)	G (0.021)	G (0.012)	G (0.013)	NA	NA
	Vogtle 1	No Data	No Data	No Data	G (2)	G (2)	G (3)	G (3)	G (0.006)	G (0.004)	G (0.002)	NA	NA
	Vogtle 2	No Data	No Data	No Data	G (2)	G (1)	G (2)	G (3)	G (0.006)	G (0.003)	G (0.006)	NA	NA
Average (NEI)	D	G	G	W (1Q)	G (3)	G (3)	G (3)	G (6)	W (0.027)	G (0.013)	G (0.012)	NA	NA
	E	G	G	W (1Q)	W (4)	G (3)	G (2)	W (9)	W (0.027)	G (0.008)	W (0.024)	NA	NA
Average (NRC)	Davis-Besse	No Data	No Data	No Data	G (1)	G (0)	G (2)	G (5)	G (0.004)	G (0.008)	G (0.005)	NA	NA
	Pt Beach 1	No Data	No Data	No Data	G (2)	G (1)	W (12)	G (3)	Y (0.051)	G (0.003)	G (0.013)	NA	NA
	Pt Beach 2	No Data	No Data	No Data	G (1)	G (2)	W (10)	G (3)	Y (0.051)	G (0.003)	G (0.011)	NA	NA
	TMI 1	G	G	G	G (1)	G (1)	G (2)	G (4)	G (0.021)	W (0.018)	G (0.004)	NA	NA
Declining (NEI)	F	G	G	G	W (5)	W (4)	G (3)	W(14)	G (0.016)	W (0.027)	G (0.010)	NA	NA
	G	G	W (1Q)	G	G (2)	G (1)	G (1)	W(22)	G (.0.016)	G (0.010)	G (0.019)	NA	NA
	H	W (5Q)	W (1Q)	G	G (3)	G (2)	G (5)	G (4)	G (0.012)	G (0.001)	G (0.003)	NA	NA
Declining (NRC)	D C Cook 1	No Data	No Data	No Data	G (2)	G (1)	W (6)	G (2)	G (0.014)	G (0.007)	G (0.010)	NA	INS
	D C Cook 2	No Data	No Data	No Data	W (4)	G (0)	W (6)	G (1)	G (0.014)	G (0.006)	G (0.008)	NA	INS
	Cooper	No Data	No Data	No Data	G (1)	G (1)	W (11)	G (2)	G (0.011)	G (0.029)	NA	G (0.014)	PI
	Hope Creek	G	G	No Data	W (5)	G (2)	W (7)	G (7)	G (0.018)	W (0.042)	NA	G (0.006)	PI

Table 2. Benchmarking Summary

Performance Category	Plant	RCS -Activity[1]	RCS Leakage[1]	Contain-ment Leakage[1]	Scrams[2]	SSAs[2]	SSFs[3]	Transients >20%[4]	Safety System Performance Indicator (Unavailability)				Method of Problem Identification
									EAC[5]	HPI[6]	AFW[7]	RHR[8]	
Watch List (NEI)	I	No data	G	G	G (1)	G (2)	W (9)	G (8)	G (0.019)	G (0.015)	NA	W(0.017)	PI
	J	No data	G	G	W (4)	G (2)	W (9)	G (7)	G (0.019)	G (0.018)	NA	G (0.011)	PI
	K	G	G	G	G (3)	G (3)	W (8)	G (7)	G (0.012)	G (0.021)	NA	G (0.009)	PI
	L	G	G	G	W (4)	G (3)	W (6)	G (4)	G (0.012)	G (0.029)	NA	G (0.008)	PI
	M	G	G	G	W (5)	G (3)	W (11)	W (26)	G (0.014)	G (0.009)	G (0.009)	NA	INS
	N	G	G	G	G (2)	G (2)	W (11)	W (23)	G (0.014)	G (0.010)	G (0.005)	NA	INS
	O	G	W (1Q)	No data	G (1)	G (0)	W (8)	W (13)	W (0.026)	W (0.071)	NA	G (0.006)	PI
	P	G	G	No data	W (5)	G (1)	W (13)	W (14)	W (0.026)	G (0.037)	NA	G (0.001)	PI
Watch List (NRC)	Millstone 1	No data	No data	No data	G (1)	G (1)	W (21)	G (7)	G (0.016)	G (0.012)	NA	G (0.005)	INS
	Millstone 2	No data	No data	No data	W (5)	G (2)	W (17)	G (7)	G (0.013)	G (0.007)	G (0.010)	NA	INS
	Millstone3	No data	No data	No data	G (1)	G (1)	W (21)	G (5)	G (0.016)	G (0.013)	G (0.020)	NA	INS
	Maine Yankee	No data	No data	No data	G (2)	G (0)	W (6)	W (12)	No data	No data	No data	No data	INS
	Indian Pt 3	G	G	No data	G (2)	G (2)	W (14)	W (9)	W (0.026)	G (0.003)	G (0.006)	NA	INS
	Crystal R 3	No data	No data	No data	G (1)	G (2)	W (9)	G (8)	G (0.009)	G (0.012)	G (0.007)	NA	PI

1 Where indicator has entered the white or yellow band, the number of quarters within those bands is indicated.
2 Values in () are the maximum number of events within a calendar year over the period 1993 to 1997; performance assessed using thresholds of 3 (g-w), 6 (w-y), and 25 (y-r)
3 Values in () are the maximum number of events within a calendar year over the period 1993 to 1997; performance assessed using a threshold of 5
4 Values in () are the maximum number of events within a calendar year over the period 1993 to 1997; performance assessed using a threshold of 8
5 Performance assessed using thresholds of 0.025 (g-w), 0.050 (w-y), and 0.10 (y-r)
6 Performance assessed using thresholds of 0.015 for PWRs and 0.04 for BWRs (g-w), 0.050 for PWRs and 0.12 for BWRs (w-y), and TBD for PWRs and 0.50 for BWRs (y-r)
7 Performance assessed using thresholds of 0.02 (g-w), 0.06 (w-y), and 0.12 (y-r)
8 Performance assessed using thresholds of 0.015 (g-w), 0.05 (w-y), and TBD (y-r)

Table 3 Year of Occurrence of Declining Performance Indications

		Barrier PIs			Initiating Events PIs		Mitigating Systems PIs				
								Safety System Performance Indicator (Unavailability)			
Performance Category	**Plant**	**RCS Activity**	**RCS Leakage**	**Contnmt Leakage**	**Scrams**	**Transients >20%**	SSFs	**EAC**	**HPI**	**AFW**	**RHR**
Average (NEI)	D			98				97, 98			
	E			95	94	94		97, 98		95 - 97	
Average (NRC)	Pt Beach 1						97	93 - 97			
	Pt Beach 2						97	97			
Declining (NEI)	F				94	94, 95			95 - 98		
	G		96			93, 95					
	H	95 - 98	96								
Declining (NRC)	D C Cook 2				95		97				
	Hope Creek				94		96, 97				
Watch List (NEI)	I						93, 94, 96				95
	J				95		93, 95				
	L				94		93				
	M				94	93 - 95	95, 96				
	N					93 - 95	95, 96				
	O		98			93	93, 94	93 - 95	93 - 96		
	P				94	93, 94	93, 96	93 - 95			
Watch List (NRC)	Millstone 2				93		93 - 97				
	Maine Yankee					93	95				
	Indian Pt 3					96	93, 97	98			

1. Do the PIs as a set differentiate between superior, average, declining trend, and watch list plants?

   The set of PIs do differentiate between watch list and other plants in both the number and magnitude of the PIs. In Table 2, the watch list plants typically have two or three PIs well into the white band. Superior performers have at most one PI in the white band, and average or declining plants have two PIs relatively low in the white band. It is difficult, however, to differentiate between the NEI declining trend plant F and some of the watch list plants on the basis of the PIs alone. Table 3, which shows the concurrence of the PIs, provides additional insights. For the average plants, there were four cases where a cornerstone had two white (or yellow) PIs and two cases where two cornerstones each had one white PI. For the declining trend plants, there were two situations when a cornerstone had two white PIs and one instance when two cornerstones each had one white PI. Of the 14 watch list plants, there were two cases where one plant had three white PIs in the same cornerstone and four other times when other plants had two white PIs in the same cornerstone. In addition, there were seven times when two cornerstones each had one or more white PIs. Plant I, while it had no concurrent white PIs, did have SSFs in the white band for two years before and one year after the RHR SSPI was in the yellow band. It should be noted that there were several cases where inspection findings could have impacted the performance assessment of declining trend and watch list plants, and they are not included in the above analysis.

2. How effective are individual PIs at differentiating between plants with different levels of performance?

   The most effective PIs for differentiating between plant performance are SSFs and Transients, largely because they have the most non-zero data points and are considered to be more leading indicators to SSPIs and reactor scram indicators, respectively. In the Arthur Andersen statistical analysis of the existing NRC PIs, SSFs correlated highly with the discussion plant list. It is the most consistent of the proposed PIs in identifying the watch list plants. SSFs provided additional information not available in the other PIs. All of the watch list plants and four of the seven declining trend plants were in the white band, while only two average plants and no superior plants were in the white. In the case of Transients, six watch list plants, two declining trend plants, one average plant, and one superior plant were in the white. The barrier PIs were not available for all plants. Of those plants with data, one of four superior plants was in the white band, along with two of three average plants, two of three declining trend plants, and one of eight watch list plants. The Scram PI was in the white band for no superior plants, one of six average plants, three of seven declining trend plants, and five of fourteen watch list plants. The SSA PI was in the green band for all plants except one declining trend plant that was in the white band; that plant was also in white band in Scrams, Transients, and the high-pressure injection SSPI. The SSPI was out of the green band for none of the superior plants, five of six average plants (including two plants in the yellow band), two of seven declining trend plants, and four of the thirteen watch list plants for which the data were available.

3. Do the PIs demonstrate timely response (i.e., do not go directly from green to red)?

   Only two of the plants we benchmarked, Point Beach Units 1 and 2, went into the yellow band between 1993 and 1997. This occurred in just one indicator, the Emergency AC (EAC) PI. The indicator was in the white band from the first quarter of 1993 (when the SSPI started) through the second quarter of 1996. It was in the green band from the third quarter of 1996 through the first quarter of 1997, then went into the yellow band in the second quarter of 1997. Although the PI

went directly from the green band to the yellow, the NRC would have had substantial opportunity to engage the licensee on this issue in prior periods
when the PI showed early declining trends. There were no instances in which a PI was in the red band for any of the plants analyzed.

4. Do the PIs show declining trends prior to SMM actions? If so, which ones are most effective? If not, would they be expected to show a declining trend?

The staff looked at the PIs for each of the selected watch list plants in the year prior to its going on the watch list. The only PIs that entered the white band in that year were SSFs for 6 of the 14 plants, and Transients for 3 plants. A sensitivity analysis, in which the threshold was adjusted up and down by 1 or 2, produced the results shown in Table 4. Moving the threshold down (lowering it) would be expected to capture additional plants, while moving it up (raising it) would be expected to capture fewer plants. The table shows the total number of plants captured as the thresholds are changed. Of the 14 watch list plants selected for this analysis, 7 of them had problems that should be identified by PIs - plants I, J, K, L, O, P, and Crystal River 3. In the year before going on the watch list, the SSF indicator went into the white band for plant J and Crystal River 3, and the Transients indicator went into the white band for plants I and J. By lowering the SSF threshold from 5 to 4, plants L and O would be captured. Lowering the SSF threshold to 3 would pick up plants I and K. Changing the Transients threshold would not capture any additional watch list plants. Plants M and N and Millstone 1 and 2 went into the white band for SSFs, although we would not have expected their problems to be identified by PIs. Maine Yankee was in the white band for Transients, although we did not expect the problems there to be identified by PIs.

Table 4. Watch List Plant Sensitivity Analysis

	-2	-1	0	+1	+2
SSFs	12	8	6	5	4
Transients	4	3	3	3	2

5. Do the PIs show declining trends prior to ASP events? If so, which ones are most effective?

There were 11 significant (>10^{-5} CCDP) ASP events in 1995 through 1997 affecting 13 units. For each affected plant, the staff looked at the PIs in the year prior to the event. The ASP event at Comanche Peak was a reactor trip with AFW unavailable, and the Unplanned Scrams (4) and Transients (11) at Unit 2 were in the white band. This would have provided the NRC with an opportunity to look into the licensee's performance prior to the ASP event. However, the AFW SSPI was near the middle of the green band. There were three other ASP events where any of the PIs were in the white band the year prior, and in each case it was the Transients indicator only. Two of the events involved EDG failure or unavailability, with the EAC SSPI well into the green band. The third event was an HPI line leak. In the two other ASP events involving reactor trips, each licensee had just one scram the year prior and no other PIs in the white band. In the four other events in which one train of AFW was unavailable, the AFW SSPI at each plant was in the green band and no other PIs in the white band. The other ASP events included two LOOPs with no PIs in the white band.

6. How well does the set of PIs conform to those selected by Arthur Andersen for use in the trending methodology currently being used in the SMM process?

The staff looked at the correlations of the current PIs to the SMM discussion plants that was performed by Arthur Andersen. Those indicators that had a high correlation included scrams, SSFs, forced outage rate (FOR), equipment forced outages per 1000 critical hours (EFO), and several of the cause codes. The proposed PI set includes indicators that are similar to those. Scrams per 7,000 critical hours is similar to the current scrams except that it includes manual scrams as well as automatic scrams and is calculated as a rate. The Transients indicator captures FOR and EFO information. The SSF indicator is the same as the one used in the current PI program. The cause codes measure programmatic causes of events. If those programmatic weaknesses manifest themselves as degradations in performance that affect cornerstone objectives, they should result in events that are captured in the proposed PIs or risk-informed inspections.

7. Do small decreases in the green-white thresholds capture more of the watch list and declining trend plants (sensitivity analyses)?

   A sensitivity analysis, in which the green-white threshold was adjusted up and down, produced the results shown in Table 5. The Number of Plants column indicates the number of plants for which data were available for that PI. The table shows the total number of plants captured as the threshold is changed. This analysis shows that Unplanned Scrams are sensitive to small increases and decreases in the threshold, and that Transients are sensitive to small decreases in the threshold. The SSPI and Barriers indicators are essentially insensitive to small changes, with the exception of the AFW SSPI.

Conclusions

As a result of this benchmarking effort, the following conclusions were reached:

1. The SSF indicator is the most effective at identifying watch list and declining trend plants identified in the SMM process. All of the watch list plants and four of the seven declining trend plants entered the white band in SSFs some time between 1993 and 1997. NEI used SSFs as a surrogate for the SSPI. Our analysis showed that it was not particularly useful for that purpose. This may have been due to the limited number of SSPI systems, inconsistent reporting of the SSPI data from plant to plant, or limitations in our ability to use the reliability portion of the indicator. However, the SSF indicator provided information not found in the other PIs. Therefore the SSF indicator was added to the set of proposed indicators. The next most effective indicator with respect to identifying watch list plants is Transients. By their nature, SSFs and Transients are considered to be leading indicators of the more risk-important indicators, Scrams, Risk-Significant Scrams, and the SSPI. They have no Required Regulatory Response threshold because they themselves are not risk-significant. They do, however, provide the best correlation with plant performance, as defined by the SMM process, for those plants analyzed.

Table 5. Performance Indicator Sensitivity Analysis

	Number of Plants	Number Change				
		-2	-1	0	+1	+2
Scrams	33	22	13	9	5	0
SSAs	33	16	6	1	0	0
SSFs	33	21	21	20	16	15
Transients	33	18	12	10	8	8

Table 5. Performance Indicator Sensitivity Analysis

	Number of Plants	Number Change				
		-2	-1	0	+1	+2
	Number of Plants	**% Change**				
		-20	**-10**	**0**	**+10**	**+20**
EAC SSPI	32	9	7	7	2	2
HPI SSPI	32	7	5	4	3	2
AFW SSPI	23	5	3	1	1	0
RHR SSPI	9	2	2	1	1	0
	Number of Plants	**% Change**				
		-20	**-10**	**0**	**+10**	**+20**
RCS Act	18	2	2*	2	2	2
RCS Leak	20	5	3	3	2	1
Ctmnt Leak	16	2	2	2	2	2

* A 10 percent decrease in the white-yellow threshold would put both plants in the yellow band.

2. The barrier PIs provided minimal differentiation between plant performance categories as identified by the SMM process. Few plants crossed into the white zone between 1993 and 1997, and those that did included one or two in each performance category. These results were rather insensitive to the thresholds. The plants typically spiked well into the white band and returned to the green band the next quarter. The barrier PIs appear useful for specific, normally short duration problems.

3. The SSA indicator proposed by NEI did not differentiate between plants or add any new information. Only one plant, a declining trend plant, was in the white band, and it was also in the white band for Transients. Lowering the threshold by one would capture two average plants and three watch list plants, all of which were identified by other PIs. In addition, the SSA indicator did not show a strong correlation to the discussion plants in Arthur Andersen's analysis. For these reasons, we do not include SSAs in our proposed set of indicators.

4. The Scrams indicator did a pretty good job of differentiating between the performance categories, but it only identified about one-third of the watch list plants.

5. Most of the SSPIs did not show declining performance consistent with the outcomes of the SMM process. The percentage of plants categorized as average that were out of the green band was much higher than any other performance category. The only yellow bands in our set of plants occurred in the Emergency AC SSPI indicator for Point Beach 1 and 2, classified as average performers. There may be inconsistencies in the reporting of the data; this will need to be investigated further.

6. The set of PIs did identify watch list plants prior to going on the watch list to some extent (i.e. several PIs dipped into the white band). This analysis indicates that adjusting the thresholds would be necessary to better match the outcomes of the SMM process.

7. As a result of the benchmarking analyses and detailed qualitative evaluation of the PIs in each cornerstone area, the staff is confident that the overall set of PIs provides a reasonably accurate depiction of plant performance in those areas that the PIs monitor. Also ,we believe the PIs are useful for determining when additional NRC inspection is warranted.

EMERGENCY PREPAREDNESS CORNERSTONE

Background
There are three performance indicators for this cornerstone that measure the performance of the Emergency Response Organization (ERO); drill/exercise performance (DEP), the readiness of the ERO, and the Alert and Notification System availability (ANSA). Both long-term and short-term thresholds have been established for each of these PIs for both the white and yellow bands; there are no red bands for these indicators. Each of the PIs is measured in percentages.

Scope
This analysis evaluated the DEP and ANSA PIs, since data for the ERO Readiness PI were not readily available. To perform the DEP analysis, data was collected from 70 plants from 1994 through 1997. The data was compared to the 24 month threshold because data were not available in six month increments. For the ANSA analysis, siren data was obtained from 20 plants from 1995 through 1997.

Method
The PI data were analyzed to identify the lowest ERO and ANSA performance during the study period. This value was compared to the thresholds to identify the worst-case performance band.

Results
The performance of the DEP went into the white band for 9 of the 70 plants analyzed in the 4 year period. Those plants are shown in Table 6, along with the lowest value of the PI during the period. Of the 20 plants analyzed for the ANSA, only Kewaunee was in the white band, with a value of 91.1%.

Table 6. Emergency Preparedness Benchmarking Summary

	Cooper	Haddam Neck	Palo Verde	Prairie Island	Quad Cities	River Bend	TMI	WNP-2	Wolf Creek
DEP	70%	80%	70%	80%	80%	80%	80%	80%	85%

Conclusions
In 1998 the NRC identified Clinton as a plant with a large number of concerns in emergency preparedness. The DEP PI for Clinton for the past 4 years does not indicate any decline in performance. Therefore, the indicator would not have identified Clinton’s weaknesses. Review and oversight of licensee self assessments, as recommended in the baseline inspection program would, however, have likely identified these concerns. In contrast, the DEP PI for Three Mile Island appropriately shows the decline in performance that was identified in 1997 during NRC inspections. Cooper was also identified as a plant with performance problems and the PI shows this status. In

general, the plants identified by this analysis were consistent with those identified as having a deteriorating trend in EP performance.

Tests of individual plant performance against the 6 month threshold have not been performed due to the lack of sufficient plant-specific data in any 6 month interval. It is believed that sufficient data will be available in the future to validate the short-term threshold. The 6 month threshold could be validated after a year of implementation. This will also give an opportunity to revisit the 24 month threshold.

Evaluation of availability data for ANS systems shows the historical high reliability of these systems. Very few plants experience ANS availability below the threshold value and none experienced availability below the regulatory value of 90%. One plant did show availability below the threshold and this performance clearly represents an unusually low ANS system performance.

OCCUPATIONAL EXPOSURE CORNERSTONE

Background

There is one performance indicator for this cornerstone that comprises the sum of three measures of licensee performance in controlling worker doses during work activities in elevated radiation fields or airborne radioactivity areas. The white band for the PI is either more than five occupational radiological occurrences in a rolling 3 year interval or more than two occurrences in a rolling 12 month interval. The yellow band is more than 11 occurrences in a rolling 3 year interval or more than five occurrences in a rolling 12 month interval. There is no red band for this PI.

Scope

The staff and NEI both identified sites whose performance in occupational radiation protection activities was considered to be below or declining from industry standards. The combined list totaled 14 sites. The staff also identified 12 sites considered to be good performers in occupational radiation protection activities. NEI provided data from 1996 through 1998 on 9 of the 14 poor performers and 7 of the 12 good performers. The plants were identified by numbers and not by plant names. The staff also collected the SALP categories in Plant Support for these plants, since plants with a 2 or 3 in that functional area normally have poor radiation protection programs.

Method

The PI data for the 16 plants was analyzed by the staff to compare the highest PI values (both 3 year and 1 year totals) to the thresholds and to identify the corresponding performance band.

Results

The results are shown in Table 7. The table indicates the plant number, the highest PI value during the study period, the performance band corresponding to that total, and the Plant Support

Table 7. Occupational Exposure Benchmarking Summary

	Poor Performers									Good Performers						
	1	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**16**	**19**	**20**	**22**	**23**	**24**	**25**
SALP Score	3	2	2	2	2	2	2	2	2	1	1	1	1	1	1	1

	Poor Performers									Good Performers					
PI Total	W (4)	G (0)	G (1)	G (0)	W (3)	W (5)	G (2)	W (3)	W (3)	G (0)	G (1)	G (0)	G (1)	G (0)	G (2)	G (1)

SALP category at the time of the highest PI value. The PI values shown are the greatest 12 month total; only plant 8 exceeded the 3 year threshold (6 occurrences), and it was also identified by the 12 month threshold.

Conclusions
The benchmarking analysis showed reasonable agreement with the perceived performance of the plants. The plants considered to be good performers had Plant Support SALP 1s and generally low PI values; five of the nine plants considered to be poor performers had PIs in the white band and all had SALP categories of 2 or 3. (Plants number 7 and 11 were identified as poor performers by NEI but not by the NRC.) The alignment of NRC staff and Plant Support SALP categories with the performance indicator thresholds supports their initial use in assessing licensee performance.

PUBLIC EXPOSURE CORNERSTONE

Background
There is one performance indicator for this cornerstone which measures the number of occurrences of offsite reportable events. The white band for this PI is either more than six events in a rolling 3 year interval or more than three events in a rolling 12 month interval. The yellow band is either more than 13 events in a rolling 3 year interval or more than 7 events in a rolling 12 month interval.

Scope
The staff and NEI both identified sites whose performance in effluent monitoring and offsite releases was considered to be below or declining from industry standards. The combined list totaled 15 sites. The staff also identified 12 sites considered to be good performers. NEI provided data from 1995 through 1997 on 11 of the 15 plants poor performers and 6 of the 12 good performers. The plants were identified by numbers and not by plant names.

Method
The PI data for the 17 plants was analyzed by the staff to compare the highest PI values (both 3 year and 1 year totals) to the thresholds and to identify the corresponding performance band.

Results
The results are shown in Table 8. The table indicates the plant number, the highest PI value during the study period, and the performance band corresponding to that total. The PI values shown are the greatest 12 month total; plants 2 and 5 exceeded the 3 year threshold (7 occurrences each), and were also identified by the 12 month threshold.

Table 8. Public Exposure Benchmarking Summary

	Poor Performers											Good Performers					
	2	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**13**	**14**	**15**	**16**	**17**	**18**	**25**	**26**	**27**

	Poor Performers											Good Performers					
PI Total	W (5)	G (2)	W (4)	W (4)	G (0)	G (0)	G (0)	W (4)	G (3)	G (0)	G (0)	G (1)	G (1)	G (2)	G (0)	G (2)	G (0)

Conclusions

The benchmarking analysis showed some agreement with the perceived performance of the plants. The plants considered to be good performers had generally low PIs and none of them entered the white band; 4 of the 11 plants considered to be poor performers had PIs in the white band. From subsequent review and discussion of the effluent monitoring LERs, the expert panel verified that not all events reportable to the NRC in accordance with the ODCM or RETS were included in the data submitted for benchmarking. Following receipt of additional reportable data from semiannual effluent reports these threshold values will be reviewed further.

PHYSICAL SECURITY CORNERSTONE

Background

There are four performance indicators proposed for this cornerstone, two to measure the effectiveness of the Physical Protection System and two to measure the effectiveness of the Access Authorization System. The two measures of the Physical Protection System are the percent availability of security equipment for the protected area and for the vital area respectively. The thresholds have been developed based on the professional judgment of the NRC, NEI, and industry peer working group on what is the appropriate level of performance. The thresholds that are proposed are 95 to 100% availability for the green band, 85 to 94% for the white band, and below 85% for the yellow band. The two measures of the Access Authorization System are the number of reportable events involving access control and fitness for duty. The thresholds for determining acceptable implementation of these programs were developed based on the professional judgement of the NRC, NEI, and industry peer working group. The thresholds that are proposed are 0 to 2 reportable events per year in either area for the green band, 3 to 5 reportable events per year for the white band, and 6 or more per year for the yellow band.

Scope

Industry historical data was collected in an attempt to benchmark the Physical Protection System PIs. However, because of the lack of consistency in past data collection and categorization processes, valid benchmarking was not possible. An attempt was made to benchmark the Access Authorization System PIs using data from 1996 and 1997. Data for the access control PI were available but fitness for duty reportable events could not easily be separated.

Method

Access control reportable events during 1996 and 1997 were collected for nine plants that were categorized by the peer group as either poor or good performers. The worst performance band for each plant during the 2 year period was then identified.

Results

The results of the benchmarking analysis are shown in Table 9. The table indicates the worst performance band and the corresponding number of annual events during the study period.

Conclusions

The access control data generally confirmed the perception that overall, these programs were working as intended and did identify one program with known weaknesses. However, because of a lack of consistency in the criteria used for reporting the data, complete confidence in the benchmarking process was not possible. Standardization of the data reporting will be addressed during the verification and validation process.

Table 9. Physical Security Benchmarking Summary

	Good Performers						Poor Performers		
	Dresden	Harris	Palo Verde	Perry	San Onofre	Wolf Creek	River Bend	St. Lucie	Water-ford
Access Control	G (1)	G (2)	G (2)	G (1)	G (1)	G (1)	G (2)	G2	W (4)

The collegial decision on the thresholds by the industry group will be subject to review after data has been gathered for a period of time and some history is established. The thresholds should be reviewed after a 2 year period to evaluate their validity and to make adjustments if necessary.

Appendix J - Team Charter and Roster

Team Charter

PURPOSE

The purpose of the taskforce is to develop details of the framework for a more objective, risk-informed, performance-based approach to licensee performance assessment and related bases for inspection activities. Information developed as part of the task will be used in the development of risk-informed baseline inspection and performance assessment tasks.

SCOPE

This activity includes: articulation of the principals, bases, and logic of the framework; identification and evaluation of performance indicators (PIs) and associated performance thresholds for initial implementation of the framework; and determining the limitations of PIs used for performance assessment and developing inspection bases for rebaselining the inspection program. The work of the taskforce will follow and build on the defining principals and cornerstone development effort that was begun at the Performance Assessment Workshop held September 28, 1998 through October 1, 1998. It is recognized that this program will evolve and be refined over a period of years. Therefore, the intent the taskforce is to develop sufficient detail to allow the Commission to make a decision on the efficacy and direction of this new approach to licensee oversight and, if approved, lay the groundwork for initial implementation.

PRODUCT

By November 25, 1998, the taskforce will provide to the Director, Division of Inspection and Support Programs, NRR and the Director of NRR, documents describing the overall framework, performance indicators and thresholds, and related bases for the inspection program. These documents will contain the principles, bases, logic, and supporting technical information and will be in the form of appendices to a Commission Paper.

Team Roster

Patrick Baranowsky, Team Leader, AEOD
Douglas Coe, NRR
John Flack, RES
Donald Hickman, AEOD
Jeffrey Jacobson, NRR
William Johnson, RIV
George Kuzo, RII
Scott Morris, RI
Robert Palla, NRR
Gareth Parry, NRR
Ann Ramey-Smith, RES
William Ruland, RI
Gregory Smith, RI
Randolph Sullivan, NRR
John Wilcox, NRR

NRC

NUCLEAR POWER REACTOR

BASELINE INSPECTION

PROGRAM

December 1998

Developed by:

Bruce S. Mallett, Region II, Team Leader
Steven R. Stein, NRR, Assistant Team Leader
Rudolph H. Bernhard, Region II
Jin Chung, NRR
Thomas Dexter, Region IV
John Flack, RES
Ronald Lloyd, AEOD
Sam Malur, NRR
Peter Prescott, AEOD
Larry Scholl, Region I
Nick Shah, Region III
Mel Shannon, Region II
Randolph L. Sullivan, NRR
Barry Westreich, OE

Attachment 3

Table of Contents

Tables

Appendices

EXECUTIVE SUMMARY

As a part of the NRC's response to issues raised by stakeholders, internal and external to the NRC, the Commission directed the staff to develop a new regulatory oversight process. The objective was to develop a more risk-informed, efficient, and effective baseline inspection program and plant performance assessment process. To accomplish this objective, the work was divided into three projects: (1) technical framework, (2) baseline inspection program, and (3) assessment process. The technical framework task group developed the concept of *cornerstones of safety*, which are areas of reactor functions or reactor licensee activities that must be performed to a certain set of objectives to ensure that the NRC's mission is met. The inspection task group used the cornerstone framework in developing the NRC power reactor baseline inspection program.

The inspection task group comprised 14 individuals from multiple NRC offices, including each region and the Offices of Nuclear Reactor Regulation, Nuclear Regulatory Research, Analysis and Evaluation of Operational Data, and Enforcement. The group included managers and staff with experience in designing and conducting inspections of nuclear power reactor facilities.

This document is the product of the inspection task group and it describes the key concepts and process for a *risk-informed baseline* inspection program. The methodology used by the task group and the key concepts of the program are summarized below:

<u>Development Methodology</u>

The baseline inspection program was developed using a risk-informed approach to determine a comprehensive list of areas to inspect, inspectable areas, within each cornerstone. These are listed in Table 1 in Section 1.1. "Risk-informed" means that the areas were selected based on their significance from a risk perspective (i.e., they are needed to meet a cornerstone objective as derived from a combination of probabilistic risk analyses insights, operational experience, deterministic analyses insights, and regulatory requirements.

Basis documents were created to describe the scope of each inspectable area and justify inspection on the basis of risk information. The basis documents also were used to indicate whether the inspection should be designed to complement or supplement a performance indicator (Part 1 of the program) or designed only to verify a performance indicator (Part 2 of the program). Risk information matrices were developed, with input from the Office of Research, to serve as guides in planning and conducting inspections as described in Section 1.3.

The team benchmarked the concepts for the baseline inspection program with the Environmental Protection Agency and the Federal Aviation Administration. The purpose was to glean insights into how these agencies incorporated risk into their inspection programs.

Throughout the project, the inspection task force took suggestions on issues to address and comments on proposed program concepts from licensees, the Nuclear Energy Institute, and internal NRC stakeholders. The baseline inspection program document addresses these issues and incorporates the comments.

Program Overview

The power reactor baseline inspection program defines the planned activities to evaluate licensee performance at a minimum level of NRC effort over a 12-month period. The overall objective of the program is to monitor all power reactor licensees at a defined level of effort to assure licensees' performance meets the objectives for each cornerstone of safety. These cornerstones support the agency's performance goals in the NRC's Strategic Plan.

A key principle in the program is that all areas where there is a need to inspect a licensee's performance be defined as "inspectable areas". These areas are then categorized into three types of inspections to reduce inspection effort where licensee performance to meet a cornerstone objective is adequately gauged by performance indicators. The first type of inspection is termed "complementary," and is used for cornerstone areas where a performance indicator has not been established. The second type of inspection is termed "supplementary," and is used for cornerstone areas where the performance indicators provide only limited indication of performance. The third type inspection is termed verification, and is used for cornerstone areas where the performance indicator sufficiently measures performance of the cornerstone objective. In the verification type, the inspection need only verify that the performance indicator is providing the intended data. The inspectable areas, along with the type of inspection proposed for each, are listed in Section 1.1.

Another important principle in the baseline program is that each inspectable area have a basis document, which describes the scope of the inspectable area and explains why the area is included in the baseline program. Reasons may be that (1) the area is linked to the NRC's mission, (2) the inspectable area involves a key attribute of a cornerstone of safety, and (3) risk information justifies including the area in the baseline inspection program.

A third principle in the program is that the regional managers and inspectors plan the type and number of activities to inspect each year for each reactor site, based on the guidance contained in the risk information matrices (RIMs) in Section 6 and Appendix III.

Risk has been factored into the baseline inspection program in four ways: (1) inspectable areas are based on their risk importance in measuring a cornerstone objective, (2) the inspection frequency, how many activities to inspect, and how much time to spend inspecting activities in each inspectable area are based on risk information in a RIM, (3) selection of activities to inspect in each inspectable area is based on the use of a RIM, modified by plant-specific information, and (4) inspectors are trained in the use of risk information.

The baseline inspection program will be implemented by resident and regional inspectors. The actual inspection of inspectable areas will be performed by these two inspection groups using the eight procedures specified in Section 4.

Future Development Work

The baseline inspection program introduces concepts that are new to the agency's power reactor inspection program and that were developed over a relatively short time. The inspection task group recognizes that additional effort is needed before the program is complete enough for a pilot program. The following tasks were identified by the task group as necessary to complete the development of the proposed risk-informed baseline inspection program:

- Write and test the inspection procedures for the baseline inspection program.

- Train resident and regional inspectors in how to apply risk through the baseline inspection program, how to implement the new program, and how to use the new inspection procedures.

- Instruct regional managers in the new concepts for the baseline program and the new methods for planning baseline inspections.

- More completely benchmark the baseline inspection program against past plant performance.

- Additionally review the inspectable areas after fully verifying the PIs to make adjustments in inspection scope and level of effort.

- Review the inspectable areas against regulations to identify any areas that are not currently addressed in the regulations.

- Continue to review stakeholders' comments on and concerns with the proposed baseline inspection program.

- Establish a process for modifying the baseline inspection program as changes are made to the performance indicators.

- Further develop the process for continually evaluating the baseline inspection program and feeding back lessons learned.

- Determine the type of inspection (resident, regional specialist, team) to be used in each part of the program and in each inspectable area.

8 PROGRAM PART 1: INSPECTABLE AREAS

8.1 Inspectable Areas Required in the Baseline Program

The baseline inspection program requires that the inspectable areas in Table 1 below be reviewed at each nuclear power plant each year. The inspectable areas verify aspects of key attributes for each of the associated cornerstones, and their link to the attributes they are measuring is depicted in the cornerstone charts in Appendix II. The inspectable areas are characterized as receiving one of three types of cornerstone inspections: complementary inspection, supplementary inspection, or verification inspection. A complementary inspection verifies performance in areas that are not measured by a performance indicator; a supplementary inspection augments information from performance indicators that do not sufficiently measure performance in a cornerstone area; and a verification inspection verifies the accuracy and completeness of the data used as the basis for a performance indicator used to fully measure performance of a cornerstone area. Therefore, the amount of inspection effort within each inspectable area is depends on the applicability of a performance indicator to the cornerstone area. Table 1 identifies the type of inspection received by each inspectable area in the seven cornerstones.

Table 1: Inspectable Areas by Cornerstone

- <u>Initiating Events Cornerstone</u>

  Adverse weather preparations (complementary)
  Equipment alignment (supplementary)
  Emergent work (complementary)
  Fire protection (complementary)
  Flood protection measures (complementary)
  Heat sink performance (complementary)
  Identification and resolution of problems and issues (complementary)
  Inservice inspection activities (complementary)
  Maintenance rule implementation (supplementary)
  Maintenance work prioritization and control (supplementary)
  Nonroutine plant evolutions (supplementary)
  Piping system erosion and corrosion (complementary)
  Refueling and outage activities (complementary)

- <u>Mitigating Systems Cornerstone</u>

  Adverse weather preparations (complementary)
  Changes to license conditions and safety analysis report (complementary)
  Emergent work (complementary)
  Equipment alignment (supplementary)
  Fire protection (complementary)
  Flood protection measures (complementary)
  Heat sink performance (complementary)
  Identification and resolution of problems and issues (complementary)
  Inservice testing of pumps and valves - ASME Section XI (complementary)
  Licensed operator requalification (complementary)
  Maintenance rule implementation (supplementary)
  Maintenance work prioritization and control (supplementary)
  Nonroutine plant evolutions (supplementary)
  Operability evaluations (complementary)
  Operator workarounds (complementary)
  Permanent plant modifications (complementary)

Table 1: Inspectable Areas by Cornerstone (continued)

Post maintenance testing (supplementary)
Refueling and outage activities (complementary)
Safety system design and performance capability (complementary)
Surveillance testing (supplementary)
Temporary plant modifications (complementary)

- Barrier Integrity Cornerstone

Changes to license conditions and safety analysis report (complementary)
Equipment alignment (supplementary)
Fuel barrier performance (verification)
Identification and resolution of problems and issues (complementary)
Inservice inspection activities (complementary)
Large containment isolation valve leak rate and status verification (verification)
Licensed operator requalification (complementary)
Maintenance rule implementation (supplementary)
Maintenance work prioritization and control (supplementary)
Nonroutine plant evolutions (supplementary)
Permanent plant modifications (complementary)
Refueling and outage activities (complementary)
Surveillance testing (supplementary)
Temporary plant modifications (complementary)

- Emergency Preparedness Cornerstone

Alert and notification system testing (verification)
Drill and exercise inspection (verification)
Emergency action level changes (complementary)
Emergency response organization augmentation testing (complementary)
EP training program (verification)
Identification and resolution of problems and issues (complementary)

- Occupational Exposure Cornerstone

Access control to radiologically significant areas (supplementary)
ALARA planning and controls (complementary)
Identification and resolution of problems and issues (complementary)
Radiation monitoring instrumentation (complementary)
Radiation worker performance (complementary)

- Public Exposure Cornerstone

Gaseous- and liquid- effluent treatment systems (supplementary)
Identification and resolution of problems and issues (complementary)
Radioactive-material processing and shipping (complementary)
Radiological environmental monitoring program (complementary)

- Physical Security Cornerstone

Access authorization (supplementary)
Access control (complementary)
Changes to license conditions and safety analysis report (complementary)

Identification and resolution of problems and issues
Physical protection system (verification)
Response to contingency events (complementary)

8.2 Basis Documents

Each inspectable area is described in a basis document (see Appendix I). The document discusses the scope of inspections within the areas and why each area needs to be inspected in the baseline program. The basis includes risk insights (from generic risk analyses and studies), analyses of significant precurser events, and the judgment of an expert panel of inspectors and risk analysts.

The basis document for each inspectable area also identifies whether a performance indicator applies to the area and how the indicator affects the inspections within the area. Inspections are either complementary, supplementary, or verification. (See Section 1.1 above.)

Inspectors will use the basis documents to focus their baseline inspections on the more risk-significant aspects of the inspectable areas, aspects that directly support the desired results and promote the important attributes of the cornerstones of safety.

8.3 Inspection Planning

The planning and tracking of hours for the baseline inspection program will be based on the total hours allocated for each cornerstone (listed in Table 3 in Section 8). Those hours were derived from a risk information matrix (RIM) that was developed to assist in applying risk to the planning and conduct of baseline inspections. (See Section 6 below and Appendix III.) RIM 1, from which the total hours were derived, establishes a baseline level of effort for each inspectable area. However, planning for monitoring of the baseline inspection program will be based on the cornerstones and not the individual inspectable areas.

Each year, inspections at each nuclear power reactor site will be planned using a process much like the current Plant Performance Review process. The primary steps in the planning process will be:

(1) In each cornerstone, determine the inspectable areas applicable for the specific plant for the upcoming period from Table 1 in Section 1.1 of this document. The only reason to exclude an inspectable area is that no activities in the area are expected to occur at the site during the next 12 months (e.g., refueling and outage activities).

(2) For each inspectable area, select the systems or activities to be inspected during the next 12 months. Generic guidelines for selection of the type of system or activity can be obtained from RIM 2. Modifications to these generic guidelines should be made to accomodate site-specific information. Guidelines for the number of systems or activities can be obtained using RIM 1.

(3) Schedule inspection of the systems and activities selected in steps 1 and 2 above over the next 12 months.

The *total* hours for inspecting the systems and activities for all inspectable areas within a cornerstone are fixed for the 12-month period and must not exceed the hours specified in Section 8 of this document. The planned hours for inspecting each inspectable area in a cornerstone, however, are not fixed and do not need to be tracked or to match those shown in RIM 1.

8.4 Event Followup

NRC normally follows up events in three ways: (1) events of low significance receive minimal follow up, (2) events of greater significance are follower up by a special team, and (3) events of moderate significance receive more follow up than1 and less than 2. The baseline program is designed to do an initial screening of all licensee event reports and following up only some of the more routine, noncomplex events. The program includes a procedure for event followup to be used in conjunction with inspections in the various inspectable areas. Whether to follow up other events with regional discretionary resources would depend on the significance of the event and whether the plant has been receiving only baseline inspections.

The followup of more extensive, nonroutine events is outside of the scope of the baseline inspection program and would be performed with reactive inspection resources. The decision to follow up such events would be made on a case-by-case basis by NRC regional management and as directed by senior NRC management in accordance with NRC Management Directive 8.3, "NRC Incident Investigation Procedures."

8.5 Plant Status Review

The baseline inspection program includes a procedure that is used in conjunction with other inspectable areas for routine control room and plant walkdowns of all safety-significant plant areas. For example, plant areas that contain equipment included within the scope of the maintenance rule, areas with significant radiological hazards, and areas with important physical security equipment would be included in this inspection. Inspection activities would also include a review of control room logs, observations of operator shift turnovers, and review or observation of the facility's plan-of-day meeting and management's review of plant deficiencies.

The primary objective of these inspection activities would be to ensure that the inspectors are aware of current plant and equipment problems and have an appropriate level of understanding of the risk significance of the proposed or ongoing operations, maintenance, and testing activities. The inspection activities would focus on identifying and understanding emergent plant issues, potential adverse trends, current equipment problems, and ongoing activities and their overall impact on plant risk. There would be an independent assessment of the licensee's effectiveness in entering system and component deficiencies into the corrective action program.

These aspects of the inspection effort are important because they will be used in the risk-informed selection process described in Section 1.3 to modify the scope and depth of inspections in other inspectable areas that support assessment of all cornerstone areas.

No performance indicators have been established for emerging plant issues or ongoing activities. In addition, performance indicator data is a lagging indicator and cannot reveal current status of equipment. Therefore, the baseline inspection program includes inspecting emerging plant issues, ongoing activities, and current status of equipment. The procedure to use in performing this inspection is referred to in Section 4 of this document.

Hours have not been allotted specifically for plant walkdowns, they have been factored into the hours allotted for each affected inspectable area.

9 PROGRAM PART 2: PROCESS FOR VERIFYING PERFORMANCE INDICATORS

The NRC and the nuclear power industry have established a set of performance indicators (PIs) for the seven cornerstones of safety. The NRC will evaluate licensee performance within the cornerstones using a combination of PIs and inspection information. Therefore, it is important that the NRC verify the accuracy of the data used as a basis for the indicators.

As part of the baseline inspection program, the NRC staff will periodically review the PI data to determine its accuracy and completeness and to compare the PI indication of performance to performance indicated by inspections. The NRC staff will collect and review licensee plant-specific PIs and will selectively review the objective raw data that formed the basis of the PIs.

As part of the 12-month planned inspections, the inspectors will review PI data. The baseline inspection program includes a separate inspection procedure for verifying PIs. The procedure will work in conjunction with the inspectable areas that include verifying specific PIs. The sampling of PI data will verify that (1) operating experience was entered into the licensee's PI database, (2) the data was appropriately characterized, (3) reporting thresholds were in accordance with agreed upon definitions and reporting criteria, and (4) any models used to prescribe action levels based on a PI produce acceptable results. The inspector will review any changes that licensees make to their PI databases or the models for determining action levels if the changes might alter PI results. As part of the PI verification, the inspector will determine whether PI action thresholds were appropriately set and not modified without adequate technical review.

10 PROGRAM PART 3: PROCESS FOR EVALUATING PROBLEM IDENTIFICATION AND RESOLUTION

The primary means of reducing risk is an effective problem identification and resolution program to correct deficiencies involving human performance, equipment, and programs and procedures.

In general, licensees identify problems (conditions adverse to quality) by three processes: (1) problem reports or condition reports that are initiated by plant personnel when they observe problems; (2) licensee self-assessments of individual departments (such as engineering, operations, and radiation control); and, (3) quality assurance audits. Problems identified by any of these processes are assessed by the licensee, root causes are determined, and corrective actions are implemented under a plant-wide corrective action program. At some plants, each department may have its own problem identification and corrective action program.

The process for evaluating problem identification and resolution will consist of reviewing the licensees' deficiency reporting process, self-assessments, quality assurance audits, root cause analyses of events, corrective actions, and followup to corrective actions to validate effective implementation. The NRC will review the licensee's activities in this area to verify that (1) the scope of licensees' identification and resolution programs bounds the key attributes in the cornerstones; (2) root causes of problems and issues have been properly determined and corrective actions are timely and effective; and, (3) the generic implication or extent of a condition has been considered. If the NRC's review indicates that the licensee has not been identifying and correcting problems for any of the key attributes, additional inspections in that area may be proposed.

The NRC program to review activities in this area has two parts. The first part is a review of the associated inspectable area within each cornerstone, along with the other applicable inspection areas. The procedures for this part will be included in the inspection procedure for each cornerstone. The second part is a biannual review of the overall problem identification and resolution programs across all cornerstones. The biannual review should not duplicate the inspections within the cornerstones. A separate inspection procedure will be developed for this part of the program.

NRC inspectors will use licensees' self-assessments to help direct these baseline inspections into worthwhile areas. However, licensees' self-assessments will not be used to reduce or replace baseline inspections.

11 INSPECTION PROCEDURES

Baseline inspections will be performed using the procedures listed below. Each portion of the baseline inspection program has certain procedures. The table below lists the procedures and how they apply to the program:

Table 2: Overview of Baseline Inspection Procedures

PROGRAM PART	PROCEDURE	APPLICABLE AREAS
Inspectable Area Review	Reactor Safety	Reviewing inspectable areas in the initiating events, mitigating systems, and barrier cornerstones
	Emergency Preparedness	Reviewing inspectable areas in the emergency preparedness cornerstone
	Radiation Safety	Reviewing inspectable areas in the occupational dose and public dose cornerstones
	Security	Reviewing inspectable areas in the physical security cornerstone
	Event Followup	Reviewing emergent events
	Plant Status Reviews	Walking down plant activities and systems while reviewing inspectable areas
Identification and Resolution of Problems	Identification and Resolution of Problems	Reviewing the identification and resolution of problems across cornerstones
Verification of Performance Indicators	Verification of Performance Indicators	Verifying the accuracy and completeness of information on which the performance indicators within the cornerstones are based

12 INTERFACE WITH OTHER NRC ACTIVITIES

12.1 Allegations

The baseline inspection program does not include resources for followup of allegations received by the NRC. The process for review of incoming allegations will continue to follow that described in NRC Management Directive 8.8, "Management of Allegations." However, the baseline inspection program is designed to allow followup during the planned reviews of the inspectable areas. Following up on allegations that require immediate response or extensive inspection effort is outside of the baseline program and is budgeted in the reactive inspection program.

12.2 Performance Assessment

Inspection findings and performance indicators will be used in assessing licensees' performance by the newly developed assessment process. The inspection findings from the baseline program (and other inspection findings for plants that receive more than the baseline program) will be recorded in inspection reports and collected in a document such as the Plant Issues Matrix. A level of risk significance, based on a risk scale, will be determined and documented for the findings. That the findings have some risk significance can be inferred from the risk-informed nature of the program. That is, the program has established risk-significant areas that will be inspected, and it directs the inspector to the more risk-significant systems and activities.

12.3 Enforcement

Violations identified during the conduct of the baseline inspection program will continue to be processed in accordance with the NRC Enforcement Policy (NUREG-1600). Violations will be followed up during the inspection of the licensee's process for identification and resolution of problems and issues. Inspection in this area will become increasingly more important as the enforcement policy is modified to expand the use of noncited violations (NCVs) and reduce licensee responses to Severity Level IV violations.

12.4 Training

The proposed baseline inspection program will require the development of new inspection procedures, which will be based on the risk insights from the inspectable area basis documents. The program will also have procedures based on cornerstones of safety rather than on SALP functional areas. Because of these significant changes, additional training is considered necessary for the inspection staff to successfully implement this program.

Training on the baseline inspection program would incude:

- Training on the organization and implementation of the baseline program (e.g., use of performance indicators, inspectable areas, planning);

- For inspectors in emergency preparedness, security, and radiation protection, training on the definition and use of risk insights in those inspection programs; and the use of generic and site specific PRA insights for all other inspectors; and

- For resident inspectors guidance on inspecting selected portions of emergency planning, radiation protection, and security.

This training would be in addition to those requirements already stated in Inspection Manual Chapter 1245, "Inspector Qualification Program For the Office of Nuclear Reactor Regulation Inspection Program." Resources for training are budgeted separately from the baseline inspection program.

13 RISK INFORMATION MATRICES

The risk information matrices (RIMs) are tools to be used in determining which activities, systems, or components are to be inspected in the baseline inspection program. The matrices are to be used, along with other generic and site specific information, in planning the baseline program at the beginning of each planning cycle, scheduling inspections within each inspection area, and during inspections by guiding the inspector to select the more risk-significant inspection samples.

The first table (RIM 1 in Appendix III) includes the inspection frequency, the number of activities or components to inspect, and total hours expected in the baseline program for each inspectable area. This RIM also describes the basis for these items. The data in RIM 1 was derived from IPE and IPEEE risk analysis, inspection experience, and history of problems at plants for the cornerstones in the reactor safety strategic performance area. Parallel efforts to develop resource estimates were made by NRC risk analysts and independently by two NRC contractors. The results of the efforts were compared and merged to create a risk informed estimate of level of effort.

A different approach was taken for incorporating the cornerstones from the radiation safety and safeguards areas into RIM 1. For these cornerstones, the relative-risk methodology used was based on relative frequency of occurrence and consequence of the events of interest. Level of effort was determined by ranking the frequency and consequence as either high or low. Areas with high frequency of occurrence and high consequence (high relative risk) were assigned the highest levels of oversight. Areas with low frequency of occurrence and low consequence (low relative risk) were assigned little or no oversight. Areas of low frequency of occurrence and high consequence were assigned a moderate level of oversight.

A second table, a generic risk insights table (RIM 2 in Appendix III), was developed based on reviews of IPE and IPEEE databases, summary NUREGs on the IPE results, and contractor reports dealing with risk insights. This table is to be used for generic insights for inspection planning until site-specific tables are developed.

14 DEFINITIONS

14.1 *Baseline Inspection Program*

The set of risk-informed inspectable areas that, together with performance indicators, provide sufficient information to assess licensee performance within cornerstones and to detect trends in performance. It is the minimum inspection performed at all operating nuclear power plants.

14.2 *Complementary Inspection*

Inspection within a cornerstone area for which a performance indicator has not been identified.

14.3 *Cornerstone*

The fundamental building block for the regulatory oversight process. Acceptable licensee performance in each cornerstone provides reasonable assurance that the NRC's overall mission of adequate protection of public health and safety is met.

14.4 *Deterministic Approach*

Considering a set of challenges to safety and determining how those challenges should be mitigated.

14.5 *Inspectable Area*

Those aspects of the physical plant or the licensee’s programs or processes that need to be verified to assure a desired attribute of a cornerstone is achieved or maintained so that the plant will operate safely.

14.6 *Key Attribute*

A characteristic of a cornerstone that needs to be achieved or maintained to assure public health and safety.

14.7 *Performance Indicator*

A set of data monitored over a period of time to provide a measure of licensee performance of a key attribute within a cornerstone.

14.8 *Risk Information Matrix*

A table that lists, for each inspectable area, important activities from a risk perspective, assigns a relative risk ranking for the area, suggests frequencies for inspecting the activities, and identifies a methodology for selecting risk-informed inspection samples.

14.9 *Risked-Informed Approach*

A philosophy whereby risk insights are considered together with other factors (e.g., engineering analysis and judgment, and performance history) to establish requirements that better focus attention on issues commensurate with their importance to health and safety.

14.10 *Supplementary Inspection*

Inspection within a cornerstone area for which the performance indicator is not sufficiently comprehensive to fully measure licensee performance.

14.11 *Validation (of performance indicators)*

The process of determining the degree to which a performance indicator measures performance.

14.12 *Verification (of performance indicators)*

The process of confirming the accuracy and completeness of data used as the basis for a performance indicator.

14.13 *Verification Inspection*

Inspection necessary to verify the accuracy and completeness of reported performance indicator data within a cornerstone area for which the performance indicator is sufficiently comprehensive to fully measure licensee performance.

15 PROJECTED RESOURCES

Direct inspection hours per year by cornerstone of safety have been projected for the baseline inspection program. They are given in the table below, which will be used for planning the inspections at each site over a 12-month period.

The hours are for the entire site. Sites with reactor units of different types are exceptions. At these sites there will be additional baseline inspection hours.

Table 3: Projected Resource Estimates (2-unit reactor site)

CORNERSTONE	INSPECTION HOURS PER YEAR (2-UNIT SITE*)
Initiating Events	182
Mitigating Systems	1151
Barrier Integrity	183
Emergency Preparedness	59
Occupational Exposure	123
Public Exposure	40
Physical Security	104
Total	1842

*Ror single-unit sites, the effort will be the same as shown in the table for the Emergency Preparedness, Occupational Exposure, Public Exposure, and Physical Security cornerstones. Level of effort needs to be determined for the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstones.

16 PROGRAM FEEDBACK AND ASSESSMENT

The baseline program was developed using the approach to select inspectable areas, which is different from the approach used in the existing core inspection program. In the new program inspectable areas have a risk-informed and performance-based cornerstone framework. The current program has a SALP functional area framework.

Because this is a new program and some of the data for inspection hours and inspection frequency in the new program relies on an estimate, there must be a feedback process to modify the inspectable areas, the inspection frequencies, and inspection hours. The effectiveness of the baseline program will be evaluated on a trial basis for a 12-month period. After the trial period, the baseline program will be evaluated by an expert panel composed of NRR and regional personnel. Feedback received from the participants in the trial programs will be used by the panel to improve the baseline inspection program.

It may be found that some PIs do not correlate well with plant performance or risk associated with plant activities. If so, the inspection program will be modified to provide an adequate level of inspection effort to assess licensee performance. If a PI is determined to no longer provide an indication of performance within an area, the PI will no longer be used to reduce inspection effort. In that case, the NRC will revert to inspection to assess performance in the area. As new PIs are developed, adjustments will be made in the inspection effort within the affected area and the PI will also be verified through baseline inspection program.

APPENDIX I

BASIS DOCUMENTS FOR

INSPECTABLE AREAS

December 1998

The following is an alphabetical listing of each inspectable area under the baseline inspection program. Following the list are the basis documents for each area.

INSPECTABLE AREA **Access Authorization (Personnel Screening, Fitness-for-duty and Behavior Observation)**

Scope

This area will verify that the licensee is properly implementing their personnel screening and fitness-for-duty programs, including granting, denying, and revoking unescorted access authorization into the protected area, as appropriate.

Basis

Inspection of these areas supports the Physical Security cornerstone.

This is a risk significant area because the personnel screening and fitness-for-duty processes are used to verify personnel reliability and trustworthiness prior to granting unescorted access to the site protected and vital areas and to assure continued reliability and trustworthiness after that. The establishment of reliability and trustworthiness for persons granted unescorted access to the protected area is a major component of protection against the insider threat as defined in 10 CFR 73.1 of radiological sabotage. The behavioral observation process is used to monitor the continuation of trustworthiness for persons authorized unescorted access and for escorted visitors.

An individual with malevolent intent or an individual under the influence of drugs could be granted unescorted access due to human or program failure. The frequency of this type of event has been low but the safety significance of this type event can be medium to high. The probability of a single individual causing a radiological release is low although the consequences of an individual causing a radiological release can be high depending on the individual's knowledge of plant systems.

Historically, licensees have effectively implemented the personnel screening and fitness-for-duty programs.

The licensee is required by 10 CFR 73.56 to maintain an access authorization program, which includes background investigations and psychological assessments, for granting individuals unescorted access to protected and vital areas with the objective of providing high assurance that the individuals are trustworthy and reliable and do not constitute an unreasonable risk to public health and safety including the potential to commit radiological sabotage. The licensee is also required by 10 CFR 26.10 to maintain a fitness-for-duty program that provides reasonable assurance that the workforce will perform tasks in a reliable and trustworthy manner and that they are not under the influence or impaired from any cause. Both rules require behavioral observation to detect indications of behavioral problems that could constitute a threat to public health and safety.

Performance Indicators

This area will be assessed by a performance indicator after an initial verification and validation inspection is done to confirm implementation of the program is acceptable and that reporting thresholds for a significant event meets regulatory expectations. The initial verification and validation inspection will serve to ensure valid data is used for the performance indicator. There will be a two-part PI for this area, one for access control and one for FFD. The performance indicator for this area will be based on the number and nature of reportable events. The PI data will be analyzed to determine trends. The PI provides some data regarding this inspectable area. The data provided is currently available and there are regulatory requirements to report significant events in the areas of Personnel Screening and Fitness-For-Duty. Performance indicators are established on the number of events reported to the NRC Operations Center. Reportable events per calendar year are established at 0-2 events - no inspections; 3-5 events - a baseline inspection and 6 or more events - a reactive inspection. However, the performance indicator is limited in the behavior observation area and only covers reporting issues involving supervisors and operators, who are reported for drug/alcohol problems under the fitness for duty program. Identification of these supervisors/operators may or may not have been through behavior observation identification. There are no other indicators to identify how effectively the program is being carried out in areas such as after duty hours call-outs, escort training, and manager/supervisor training. Consequently, a minimum baseline inspection should be conducted of the behavior observation program process and human performance attributes. This area should be reviewed after a 2-year period to evaluate the threshold's validity and to make adjustments as necessary.

INSPECTABLE AREA: **Access Control (Search of Personnel, Packages, and Vehicles; Identification and Authorization)**

Scope

This area will verify that the licensee has effective access controls and equipment in place designed to detect and prevent the introduction of contraband (firearms, explosives, incendiary devices) into the protected area that could be used to commit radiological sabotage and to assure that only authorized personnel are permitted unescorted access to the site protected area and vital areas. The Identification and Authorization process is to assure that, once personnel have been screened to verify their trustworthiness, those persons have a need for access and to confirm that only those persons who have been screened and have a need are granted access to the plant including vital areas. Some of the equipment involved are metal detectors, explosive detectors, x-ray machines, biometric sensors, computers, key-cards, hard keys, and card-readers.

Basis

Inspection in this area supports the Physical Security cornerstone**.**

The areas to measure are the effectiveness of the search function (personnel, packages and vehicles) and the identification and authorization. The search function is to prevent the introduction of contraband (firearms, explosives, incendiary devices) that could be used to commit radiological sabotage. The search function for detection of firearms, explosives and incendiary devices on individuals, in packages, or vehicles, is accomplished by equipment listed above or a hands-on search. The identification and authorization functions are accomplished during issuing of badges or through the use of biometrics or card-readers. The licensee must also positively control all points of personnel and vehicle access into vital areas.

The frequencies of an unauthorized individual being granted unescorted access or the introduction of contraband, described above, into the protected or vital areas are low but the consequence of risk to radiological sabotage is considered moderate.

Performance Indicators

At this time there is no performance indicator for this inspectable area that measures both equipment and human performance at the same time. The combination of both is what is used to detect and vital to the prevention of contraband entering the protected area. Meaningful tracking data on the performance of access control was not practical since much of the performance is dependent on the quality of the implementation of the tasks. The areas of search and identification and authorization will be inspected as part of the baseline inspection program. The inspection will consist of procedure reviews, self assessment reviews, observations of personnel processing, security officer performance and observation of routine testing of equipment. The same level of inspection effort would be applicable if a PI was in place. These are areas where the effectiveness of doing the task determines the effectiveness of the processes and areas where many personnel in the security organization do the tasks.

INSPECTABLE AREA: **Access Control to Radiologically Significant Areas**

Scope

This area will verify that the licensee has implemented effective Radiation Protection (RP) Barrier Integrity to prevent an uncontrolled access to an airborne, high (HRA) or very high (VHRA) radiation area that could potentially result in an exposure in excess of regulatory limits. An RP Barrier Integrity includes: identification and control of the hazard, administrative controls (RWPs, planning, procedures), physical Barrier Integrity or engineered controls (e.g., Barrier Integrity ropes, locked doors, shielding, or ventilation systems), radiological surveys and monitoring (e.g., RP technician coverage, personnel alarming dosimeter, or remote monitoring or surveillance), and radiation worker training.

Basis

Inspection in this area supports the plant facilities attribute of the Occupational Exposure cornerstone.

This inspection will review the licensee's performance in instituting the physical and administrative controls defined in Subparts G, H, I, and J of 10 CFR Part 20, applicable technical specifications (TS), and licensee procedures for airborne areas, HRAs and VHRAs and worker adherence to these controls.

Radiological risk (i.e., exposure) to a worker must be within the occupational exposure limits defined in 10 CFR Part 20 and ALARA to minimize the potential for health effects. Collectively, the access controls provide a "defense-in-depth" against a significant exposure. Industry experience has identified frequent occurrences where the failure of multiple Barrier Integrity resulted in an uncontrolled entry and, in some cases, a significant exposure.

Performance Indicators

The established performance indicator (PI) does not address airborne areas or HRAs with dose rates <1000 mrem/hr or highly contaminated areas having the potential for an exposure in excess of regulatory limits. Therefore these areas will be included in the baseline inspection. Incidents that would be tracked under this PI include:

- A single nonconformance of TS controls or comparable 10 CFR Part 20 requirements applied to all high-radiation areas (HRAs) having dose rates $\geq$ 1000 mrem/hr.

- A single nonconformance with 10 CFR Part 20 and/or licensee procedural requirements regarding radiation protection controls associated with VHRAs.

- A single occurrence of an uncontrolled exposure in excess of 10 percent (%) of the non-stochastic or 2% of the stochastic dose limits specified in 10 CFR Part 20.

INSPECTABLE AREA: **Adverse Weather Preparations**

Scope

Inspection activities in this area would focus on the effectiveness of the licensee's program for protecting mitigating systems and components from cold weather and other adverse weather related conditions. The inspection focus would be to ensure that risk significant systems and components will perform within the design assumptions for adverse weather.

Basis

Inspection of this item supports the Initiating Events and Mitigating Systems cornerstones by ensuring that the licensee takes steps to reduce the effects of weather-related initiating events and the impact of adverse weather on key portions of mitigating systems.

The inspection activities are intended to verify that the licensee has taken the necessary steps to demonstrate that the reliability, availability and functional capability of SSCs and associated components are maintained during adverse weather conditions. For example, operating experience indicates that cold weather conditions continue to cause intake structure icing, process and instrument line freezing, emergency diesel generator oil viscosity problems, essential chiller problems, and electrical problems such as grounds.

Frozen equipment can lead to a common cause/mode loss of multiple trains and loss of equipment in redundant systems without any indication of a problem until called upon to function, which would have a significant impact on plant risk. In addition, high temperature conditions can place plant equipment and systems in an unanalyzed condition, which could also have a significant impact on risk.

Performance Indicators

There are no performance indicators that have been established that can provide results related to the adequacy of the licensee's program for freeze protection or for the adequacy of the licensee's preparations for other adverse weather conditions.

INSPECTABLE AREA: **ALARA Planning and Controls**

Scope

This area will verify that the licensee maintains occupational exposure ALARA by properly planning and controlling radiologically significant work activities. Controls, as stated here, refer to those physical (e.g., locked doors, Barrier Integrity ropes, shielding, engineering controls) and administrative (e.g., surveys, planning, procedures, training, monitoring) Barrier Integrity that, in the aggregate, serve to mitigate exposure.

The focus is whether reasonable goals were established for radiologically significant work which consider previous licensee performance and industry experience, and whether the licensee's's subsequent performance met those goals. Emphasis should be placed on those jobs having a high individual and/or collective dose, being performed in an area of higher radiological risk or are of concern because of industry or licensee experience (such as spent fuel pool diving). This may include observing selected activities to verify the assumptions underlaying these goals and that the appropriate controls were implemented. The inspection should also review licensee assessments of the ALARA program to determine whether adequate administrative controls, management oversight, and exposure controls (including source term reduction) were taken.

Specific attention should be given to Planned Special Exposures and exposures to Declared Pregnant Workers, because of the inherent risk and public interest.

Basis

Inspection in this area supports the program/process attribute of the Occupational Exposure cornerstone.

This inspection will review whether the licensee meets the requirements of Subpart B to 10 CFR Part 20, which requires that a Radiation Protection program, including procedures and engineering controls, be instituted to maintain occupational dose ALARA.

Radiological risk (i.e., exposure) to a worker be within the occupational exposure limits defined in 10 CFR Part 20 and ALARA and to minimize the potential for health effects. Effective ALARA planning will ensure that adequate physical and administrative controls are in place to mitigate exposure during radiologically significant work. Industry's experience includes frequent events where problems in this area have resulted in unanticipated exposure or a loss of control of the work activity. Specific attention should be given to Planned Special Exposures, exposures to Declared Pregnant Workers, and to activities that challenge the maintenance of occupational exposure control and ALARA, such as outage and refueling planning and preparation, emergent work activities, and radiological events.

Performance Indicators

There is no performance indicator established that covers this area. Assessment of the ALARA program effectiveness is site-specific and highly dependent upon operational history, work scope, and worker experience.

INSPECTABLE AREA: **Alert and Notification System Testing**

Scope

Inspection in this area includes a review of testing activities for the Alert and Notification System (ANS) in order to assess licensee performance.

Basis:

This inspection area supports the Emergency Preparedness (EP) cornerstone and the Facilities and Equipment key attribute.

The ANS is a risk significant system for notifying the public of the need to take protective actions. The licensee generally maintains the ANS and the local governmental authorities operate the ANS when necessary. Assurance that the system has a high rate of availability increases the assurance that the licensee can protect public health and safety during an emergency. If an EP program consistently ensures that the ANS is in a high state of readiness it indicates that the program is operating at or above the threshold of licensee safety performance above which the NRC can allow licensees to address weaknesses with NRC oversight through a risk informed inspection program.

Performance Indicators

A PI, ANSA, addresses performance in this area. However, for the statistics of the PI to be valid, the testing program must be conducted in accordance with NRC guidance. The inspection verifies testing program compliance. Every site would be inspected once during the implementation of the NRC Assessment Program and thereafter if there are changes in the methodology.

Areas that would require inspection if the PI were not available include:

Review of surveillance tests for completeness
Review the disposition of a corrective actions
Review disposition of repeat items

INSPECTABLE AREA: **Changes to License Conditions and Safety Analysis Report (10 CFR 50.54 and 10 CFR 50.59)**

Scope

Inspection activities in this area focus on those changes to the facility and licensee programs performed under the requirements of 10 CFR 50.54 and those changes to the facility, procedures, tests or the Final Safety Analysis Report (FSAR) performed under the requirements of 10 CFR 50.59. The inspection activities include a review of the licensee's required submittals as specified by 10 CFR 50.54 and 50.59. A more detailed review would be performed on those changes that have the potential to be and/or appear to be intent changes. Examples of inspection areas would include safety evaluations performed by the licensee for permanent and temporary facility modifications, procedure changes, FSAR changes, emergency and security plan changes.

Basis

Inspection of this area supports the Mitigating Systems, Barrier Integrity, and Physical Security cornerstones.

Inspection of this item provides monitoring of the effectiveness of the licensee's programs for implementing changes to facility SSCs, risk significant normal and emergency operating procedures, test programs, FSAR and security plans and ensures that the changes were in accordance with the requirements of 10 CFR 50.54 and 10 CFR 50.59. This would provide assurance that the facility changes have not reduced the safety margins of the SSCs or reduced the effectiveness of the facility security plans.

Performance Indicators

No performance indicators have been established that can provide results related to the adequacy of the licensee's program for making changes to the facility.

INSPECTABLE AREA: Drill and Exercise Inspection

Scope

This Inspection area is an evaluation of licensee self assessment of performance during the conduct of drills, exercises, appropriate training evolutions and actual events. This will verify that the statistics gathered for the DEP PI represent the actual success rate of performance and provide oversight to ensure the efficacy and veracity of the licensee problem identification and resolution program as related to EP.

Basis:

This inspection area supports the EP cornerstone and the ERO Readiness, Facilities and Equipment, Procedure Quality and ERO Performance key attributes.

The implementation of the Emergency Plan is dependant on the performance of the ERO in their EP assignments. There are many areas important to Plan implementation, but the most risk significant areas of ERO performance are:

- <u>Timely and accurate classification of events</u>; including the recognition of events as potentially exceeding emergency action levels (EALs) and any assessment actions necessary to support the classification.

- <u>Timely and accurate notification of offsite governmental authorities</u>; including adequate performance of notifications as specified in the Plan.

- <u>Timely and accurate development and communication of protective action recommendations to offsite authorities</u>; including providing protective action recommendations (PARs) to governmental authorities, the decision making process to develop the PARs and any accident assessment necessary to support PAR development.

If the ERO consistently performs these activities in a timely and accurate manner, it indicates that the EP program is operating at or above the threshold of licensee safety performance above which the NRC can allow licensees to address weaknesses with NRC oversight through a risk informed inspection program.

Performance Indicators

The DEP PI has been developed to indicate performance in this area.

However, the data used to develop the DEP PI is based on licensee assessment. This inspection area verifies licensee assessment activities that generate this data.

Simulated emergency events that are identified in advance of performance as opportunities for the DEP PI would be observed. Inspection of drills and training evolutions could be unannounced, but inspection of the biennial exercise could not. The inspector would observe licensee assessment of risk significant activities and verify the determination of successes and failures. The inspector would also verify that the reported PI statistics conform to the observation.

In addition, during the biennial exercise, the licensee’s ability to identify and resolve EP related problems would be inspected, including the following areas:

- ERO proficiency in general,

- ERO ability to diagnose plant accident conditions, formulate mitigating actions and implement them under accident conditions,
- readiness and quality of EP equipment and facilities,
- direct interface with offsite authorities during exercises and drills, e.g., in the area of PAR communication and technical support,
- adequacy of communication channel testing and timely correction of communication channel deficiencies,
- adequacy of worker protection during exercises and drills, and
- ANS deficiency correction.

The licensee should identify ERO performance problems that detract from the ability to protect the public health and safety. The identification of repeat items and trends and the disposition of corrective actions would be inspected. The ability to identify and resolve problems is integral to the efficacy of an EP program. This area is meant to include any licensee efforts that assess the EP program or the performance of the ERO such as:

- self assessment reports including reports of actual events or missed classification of actual events,
- biennial exercise and drill critiques,
- audits conducted under 10 CFR 50.54(t), and
- assessments performed by the Quality Assurance organization.

If the DEP and ERO PI's are not available, the biennial exercise would have to be inspected in a broader manner. This effort would include elements in the current inspection program such as, inspection of ERO performance in the CR, TSC, OSC, EOF, damage control teams, field monitoring teams, etc. The inspection areas would include:

- Timely activation
- Facility and management control
- Analysis of plant conditions
- Classification
- Notification
- PAR development
- Dose projection
- Accident assessment
- Accident mitigation planning
- Engineering support
- Damage control
- Habitability assessment
- Site and facility personnel accountability
- Protective actions for workers
- Security activities
- Offsite monitoring teams
- PASS teams

INSPECTABLE AREA: Emergency Action Level Changes

Scope

Inspection activities in this area includes a review and assessment of changes to the Emergency Action Levels (EALs).

Basis:

This inspection area supports the Emergency Preparedness (EP) cornerstone and the Procedure Quality key attribute.

Recognition and subsequent classification of events is a risk significant activity because classification leads to activation of the Emergency Response Organization and notification of governmental authorities. However, if the EAL scheme is not in compliance with the approved configuration or with regulations, it will not result in the expected emergency classification.

Appendix E to 10 CFR Part 50, states that NRC will approve EALs. "Emergency Preparedness Position (EPPOS) on Emergency Plan and Implementing Procedure Changes," EPPOS No. 4, provides additional guidance. EAL changes are expected to be submitted for NRC review and approval prior to implementation. This inspection area addresses the need to perform this review. All changes would be reviewed.

Performance Indicators

No PI's were established that cover this area.

INSPECTABLE AREA: Emergency Response Organization Augmentation Testing

Scope

Inspection in this area involve reviews of licensee self assessments of Emergency Response Organization (ERO) augmentation including the design of augmentation tests to ensure they provide assurance that emergency response facilities could have been staffed in a timely manner if it had been necessary, the self assessment of test conduct and an analysis of test results. The inspector will also review the ability of self assessments to identify trends in results and implement the associated corrective actions.

Basis:

This inspection area supports the Emergency Preparedness (EP) cornerstone and the ERO Readiness key attribute.

The licensee system to augment the on shift staff with ERO members is a risk significant process because the ERO is critical to implementing the Plan in a timely manner. This system involves a notification system for individual ERO members, training of ERO members in its use, and testing to ensure facility activation goals can be met. The test design would be performed for every site initially and there after:

Changes to test design
Self assessment of the response
Identification of trends
A sampling of findings and the implementation of corrective actions
Disposition of repeat findings

Performance Indicators

None of the established PIs cover this area.

INSPECTABLE AREA: **Emergent Work**

Scope

The inspection activities in this area would focus on the effectiveness of the licensee's configuration controls during repair of emergent equipment failures. These inspection activities would include a review of related troubleshooting, work planning, establishment of plant conditions, tagging, conformance with Technical Specifications and restoration of equipment to service, with an emphasis on verification of plant configuration The inspection activities would be limited to risk significant emergent activities that could cause an initiating event to occur or affect the functional capability of mitigating systems.

Basis

Inspection of this item supports the Initiating Events and Mitigating Systems cornerstones.

Inspection activities are intended to verify that the licensee has taken the necessary steps to demonstrate that emergent activities are adequately planned and controlled to avoid initiating events and to ensure the continued reliability, availability and functional capability of SSCs. This would include proper control of troubleshooting, maintenance activities, and appropriate post maintenance testing. In addition, emergent failure of equipment can result in risk significant configurations if redundant equipment was already unavailable because of planned maintenance or testing.

Industry experience has shown that inadequate control of repair activities to equipment during power operation have resulted in plant transients, inoperable safety systems, and/or loss of redundancy. In addition, when the plant is at full power operation, thorough post-maintenance testing by the licensee can become more difficult and may warrant additional NRC attention to verify equipment reliability is not jeopardized due to inadequate or inappropriate testing.

In addition, the inspection activities should ensure that the licensee has appropriately considered the prioritization and timing of repairs and that the repair activities are factored in with other previously planned maintenance or surveillance activities such that overall plant risk is minimized.

Performance Indicators

There are no PIs for configuration control that provide a leading indicator of potential events or failures that could result from a failure to properly plan and control emergent work.

INSPECTABLE AREA: **EP Training Program**

Scope

Inspect training program for adequacy, changes and the knowledge level and qualifications of ERO members.

Basis:

This inspection supports the Emergency Preparedness (EP) cornerstone and the ERO Readiness and ERO Performance key attributes.

Emergency Preparedness is the final barrier in the "defense in depth" NRC regulations provide for ensuring the public health and safety. The training program must ensure that ERO members are adequately prepared to perform their assigned EP duties. The ERO members must be qualified to perform their assigned duties. The following areas would be inspected:

Review all training program modules over a six year period to ensure adequacy.
Review all changes to training program modules since the last inspection.
Review training records of a significant portion of ERO members with risk significant duties, classification, notification and PAR development, including shift management
Interview a significant portion of the above identified ERO members to verify their knowledge level
Select a sampling of other ERO positions and interview the assigned individuals for knowledge level as applicable to duties.
Review the qualifications of a significant portion of ERO members, including shift management, with duties in the risk significant areas of classification, notification and PAR development.
Review the qualifications of a sampling of other ERO members.
Review the qualifications of new members (changes) to the ERO.

Performance Indicators

Two PI's, DEP and ERO, address this area and therefore a baseline inspection is not required.

INSPECTABLE AREA: Equipment Alignment

Scope

Inspection activities in this area focus on the effectiveness of the licensee's program for changing the alignment of risk significant plant equipment . This includes changes made for operational needs and for removing equipment to/from service for activities such as maintenance, modification or testing. The inspection focus would include a review of the effectiveness of the licensee's programs for independent verification, locked valve verification, switching and tagging clearances and system lineups. The inspection activities would be more limited during power operations with increased emphasis during shutdown evolutions.

Basis

Inspection of this area supports the Initiating Events, Mitigating Systems and Barrier Integrity cornerstones.

The inspection activities are intended to verify that the licensee has an effective process for maintaining system configuration control, which ensures that the functional capability of the plant system is maintained. Systems and components that are not properly configured may not be capable of performing their intended functions, which results in a loss of availability and functional capability.

Systems or components that are not properly aligned can lead to the initiation of events, can result in personnel injuries, and can significantly impact the availability and functional capability of plant equipment, which could significantly increase the overall risk to the plant. It is understood that inspection activities will have minimal impact on reducing the frequency of initiating events. However, a review and documentation of those events does provide valuable assessment information. Inspection activities would normally be performed following emergent work activities, following risk significant system realignments, or during outage related activities.

Performance Indicators

A performance indicator for the unavailability of four systems has been identified. Due to the monitoring of a limited number of systems, this inspection supplements that PI. Also, there is no similar PI for equipment lineup during shutdown conditions, requiring this baseline inspection.

INSPECTABLE AREA: Fire Protection

Scope

The inspection includes a review of ignition sources, control of combustible materials, and fire protection systems and equipment. Fire brigade staffing, training and performance as well as equipment necessary for plant shutdown following a fire such as emergency lighting, Appendix R diesel generators (when applicable) and remote shutdown equipment would also be included as part of the inspection activities.

Basis

Inspection of this item supports the Initiating Events and Mitigating Systems cornerstones.

The inspection would review licensee controls designed to minimize the probability of a fire and would also review the availability and reliability of equipment necessary to mitigate the effects of a fire.

Proper implementation of the fire protection program is important to provide defense-in-depth against fires by maximizing prevention, detection, suppression, and mitigation capabilities for fires. An effective program reduces the risk of a fire being an initiating event. Also, in the event of a fire, reliable detection, suppression and mitigation capabilities ensure the plant can be safely shut down. Plant specific evaluations have shown internal fires to be high contributors to risk at some plants due to the potential for damaging redundant systems and multiple control circuits and due to the adverse effect on operator mitigation strategies.

Performance Indicators

There are no performance indicators that assess performance in the area of fire protection.

INSPECTABLE AREA: **Flood Protection Measures**

Scope

Inspection activities in this area focus on licensee's program to protect the plant from potential flooding. These inspection activities would include verification that compensatory measures are documented, equipment is available and staged, and equipment is routinely tested and remains fully capable to perform the intended functions. These activities would be performed at specific facilities that have the potential for external flooding and would also include those facilities with internal flooding concerns.

Basis

This activity would be an input to the Initiating Events and Mitigating Systems cornerstones.

Verification of the licensee's implementation of the flood control program would be performed to insure that the facility is capable of withstanding potential internal and external flooding. Flooding would have a significant adverse affect on the functional capability of safety and risk related equipment needed to maintain the plant in a safe shutdown condition.

Flooding has been shown to be a significant contributor to risk at some facilities. In addition, flooding has the potential to make multiple trains of equipment and support equipment inoperable which would result in a significant increase in risk to the plant. Flooding also has a significant consequence of preventing or limiting operator mitigation and recovery actions.

Performance Indicators

There are no performance indicators that have been established that can provide results related to the adequacy of the licensee's program for mitigating the consequences for flooding. Due to the rare but possibly risk significant nature of flooding events, no performance indicator was judged to be suitable for monitoring licensee performance in this area.

INSPECTABLE AREA: **Fuel Barrier Performance**

Scope

Inspection includes verification of operation of the licensee's capability and performance of in-plant radio-chemical analyses of the reactor coolant system (RCS).

Basis

Inspection of this item supports the cladding performance attribute of the Barrier Integrity cornerstone.

Inspection of fuel cladding radio-chemistry analysis performance will provide assurance that the first Barrier Integrity against release of radioactivity to the environment is maintained. Failure of fuel cladding would increase the radiation dose to workers and potentially to members of the public.

The fuel cladding Integrity is maintained by controlling reactor operation within the established operational limits. Routine sampling and radio-chemical analysis of reactor coolant will detect any fuel cladding failures. Appropriate plant procedures and measures for protecting plant workers from increased dose due to fuel failures and to prevent release of radioactivity to the environment should be implemented.

Performance Indicators

A performance indicator is provided for RCS activity. This inspectable area could be deleted, if the performance indicator for this area is acceptable and the indicator is verified.

INSPECTABLE AREA Gaseous and Liquid Effluent Treatment Systems

Scope

This area will verify that gaseous and liquid radioactive effluent treatment systems are maintained such that radiological releases are properly mitigated, monitored and assessed. The focus is to ensure that releases are reasonably controlled, that system modifications are properly performed, and that effluent and meteorological monitors are accurate and reliable. Other aspects of system operation (including administrative controls) will be assessed by reviewing licensee assessments, the Annual Environmental Monitoring Report and the Annual Effluent Release Report.

The baseline program should consist of performing in-office reviews of the Radiological Effluent Release Report to verify that the program was implemented as described in the ODCM. Additional areas of review include calibrations of the effluent and meteorological release monitors and modifications to the gaseous or liquid radwaste systems. However, the inspection should not review the original as-built liquid and gaseous system, maintenance records or administrative controls as it is expected that deficiencies in this area will be addressed through the licensee's assessment process and in the Annual Radiological Effluent Release Report. However, the baseline inspection can include walking down the liquid and gaseous release systems to observe ongoing activities (such as radwaste transfers) and to independently verify that licensee identified deficiencies were being corrected.

Basis

Inspection in this area supports the plant facilities/equipment and instrumentation and program/process attributes of the Public Exposure cornerstone.

This inspection will verify that gaseous and liquid effluent processing systems are maintained as required by General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50, Radiological Effluent Technical Specifications (RETS) and the Offsite Dose Calculation Manual (ODCM).

Radiological risk (i.e., exposure) to the public below the 10 CFR Part 20 and 40 CFR Part 190 limits and ALARA to minimize the potential for health effects. Doses below the design objectives of Appendix I to 10 CFR Part 50 are considered ALARA by the NRC. Proper operation of the effluent treatment system and monitors will ensure an adequate "defense-in-depth" against an unmonitored, unanticipated release of radioactivity to the environment. Overall industry performance has improved, but concerns still exist with abnormal releases, system modifications, and monitor operability.

Performance Indicators

The performance indicator (PI) for this item adequately addresses most aspects of the program/process and human performance attributes of this area. However, it does not address abnormal releases, system modifications, and meteorological and effluent monitor operability. The PI does address meteorological and effluent monitor calibration and setpoint verification, but a selective inspection of calibration and setpoint records is necessary to verify the integrity of the PI. Because most licensee's calibrate these monitors at a greater than annual frequency it was deemed appropriate to perform this review as part of this inspectable area rather than through the annual verification of PIs. Incidents that will be tracked as a PI include any effluent release not in accordance with 10 CFR Part 20, Appendix I to 10 CFR Part 50, ODCM, and RETS.

INSPECTABLE AREA: **Heat Sink Performance**

Scope

This inspection includes a review of the performance of normal and ultimate heat sinks. The inspection activities would focus on potential common cause failures of heat removal capabilities such as clogging of intake screens, strainers, piping and heat exchangers.

Basis

Inspection in this area supports the Initiating Events and Mitigating Systems cornerstones by ensuring initiating events are not caused by a loss of heat sink and that mitigating systems heat removal capabilities are not degraded.

The inspection would focus on events that could result in the simultaneous loss of both the normal and ultimate heat sinks due to events such as ice buildup, grass intrusion or blockage of pipes and components by other foreign materials.

Also, industry experience has shown that many plants have experienced significant problems with repeated loss of heat sink and degraded performance of heat exchangers due to problems that include corrosion, silting and fouling. Since the subject heat exchangers do not normally operate at design heat loads, it is important for the licensee to routinely monitor the performance of the heat exchangers to ensure that the heat exchangers are capable of meeting their design requirements.

Performance Indicators

None of the established PIs cover this area.

INSPECTABLE AREA: Identification and Resolution of Problems/Issues

Scope

This item will verify that the licensee has an effective problem identification and resolution program. Problem identification and resolution refers to: (1) the deficiency reporting process; (2) licensee self-assessments; and (3) Quality Assurance audits. Additionally, in some plants each department may have its own problem identification and resolution program. The focus of the inspection is on the licensee's effectiveness in identifying, resolving and preventing risk significant problems.

Basis

Inspection in this area supports all seven of the cornerstones.

The objective of this inspection is to ensure that the licensee effectively assesses performance to identify and correct situations that could impact the cornerstone objectives .

An effective problem identification and resolution program is the primary means of reducing risk by correcting deficiencies involving people (i.e., training, knowledge and skills), processes (i.e., procedures and programs), and equipment (i.e., design and maintenance) before they manifest in a significant event affecting the health and safety of workers or the public. Industry experience indicates that licensees having an effective program for identifying and resolving problems also have a reduced frequency of events.

The inspector shall select a set of outputs from a selected program for review. For each cornerstone of interest, a sample set comprising licensee assessments and deficiency reports will be selected for review. The selection will be made using information contained in the Risk Information Matrix (RIM) and insights gained from site-specific PRA results, industry experience and NRC inspection findings. Where site specific toxic hazards and grid stability problems have been identified, the resolution of these types of issues should be included in a review of corrective actions.

For selected programs, additional issues may be identified by periodic observations of specific activities such as operator simulator training, or emergency preparedness, security and fire protection drills and exercises. Some issues may also be identified by reviewing operating experience information, engineering and maintenance work request data bases, operator work around lists and the non-conformance report data base. Collectively, these issues shall also be reviewed for inclusion in the sample set.

When reviewing the sample set, consider whether individuals involved in the problem identification and resolution process effectively identify, resolve and correct risk-significant problems. Additionally determine if risk insights were used to allocate licensee resources for investigating and correcting identified deficiencies.

The inspection should verify that: (1) the licensee's assessments of problems and issues were of sufficient scope to address the key attributes of the cornerstone; (2) the risk significance of the findings was properly assessed; (3) root cause analyses and corrective actions were timely and adequate to prevent recurrence; (4) industry and NRC generic issues were considered; (5) required reports to the Commission or input to a PI were made; and (6) the performance trend indicated by the sample set was consistent with the applicable PIs.

Periodically during the inspection, discuss issues with the residents (or other inspection team members if applicable) to identify common issues that cross other cornerstones. For example, procedural adherence problems in the Occupational Exposure, Initiating Events and Barrier Integrity

Cornerstones. Review the common finding as stated above and determine if the licensee was aware of the common issues.

Additional sampling of the licensee's performance assessment feedback loop is required if: (1) recurrent issues or highly risk significant findings were identified; (2) adequate corrective actions were not taken in response to a declining trend or performance above a PI threshold; or (3) the NRC or licensee assessment results indicate risk significant findings that should have been manifested in a negative PI trend.

An observed discrepance between PI data and NRC or licensee findings is indication that additional review of PRA assumptions, re-verification of applicable PIs and an assessment of changing risk may be required.

Performance Indicators

None of the established PIs cover this area. However, some insight may be obtained from the PIs developed for each cornerstone, which may reduce the overall inspection effort in this area.

INSPECTABLE AREA: **Inservice Inspection Activities**

Scope

Inspection activities in the area would focus on the effectiveness of the licensee's program for inservice inspection, repair, and replacement of reactor coolant system (RCS) pressure retaining components. Inspection activities would include a review of the results of the steam generator tube inspections, a selected review of risk significant non-code repairs, and a review or observation of the reactor vessel ISI examinations.

Basis

Inspection activities in this area primarily support the Barrier Integrity cornerstone. Activities also support the Initiating Events cornerstone because ISI activities can detect precursors to RCS boundary failures.

The inspection activities are intended to ensure that the licensee has an effective program for monitoring degradation of reactor coolant system boundary, including steam generator tubes, control of non-code repairs to ASME components, and performing the required periodic ISI examinations.

Degradation of the reactor coolant system, steam generator tubes, or safety related support systems would result in a significant increase in risk. Degraded piping or tubes would increase the risk impact due to initiation of events. In addition, it would result in mitigating systems not being capable of performing their intended design functions. Based on these considerations, inspection activities are necessary to ensure that the licensee has an effective ISI program to ensure that risk significant degradation of the RCS boundary is identified and is promptly and appropriately corrected.

Performance Indicators

There are no performance indicators that have been established that can provide results related to the adequacy of the licensee's program for ensuring system integrity in accordance with ASME requirements

INSPECTABLE AREA: **Inservice Testing of Pumps and Valves-ASME Section XI**

Scope

Inspection activities in this area would be focused on the effectiveness of the licensee's program for testing of pumps and valves as required by ASME Section XI. Inspection activities in this area would include a review of test procedure adequacy, testing methodology, equipment trend results and observations of selected pump performance testing, valve stroke time testing, relief valve setpoint testing, and check valve testing.

Inspection Basis

Inspection activities in this area would provide input to the Mitigating Systems cornerstone.

Inspection of this area would be performed to verify that the required testing is being performed as required and that plant equipment is functioning as designed, with an emphasis on ensuring test procedures are adequate to confirm design bases requirements are being met. Section XI testing program was specifically designed to demonstrate the reliability of components and to identify degrading components prior to actual failure. The trending of the Section XI test data is necessary to identify degradation of components so that the licensee can initiate corrective actions before the degradation causes a loss of functional capability. This ensures that equipment will be available and have adequate functional capability if called upon to mitigate the consequences of an accident.

Degraded equipment, even on less significant systems can collectively have a significant impact on overall plant risk.

Performance Indicators

There are no performance indicators that have been established that can provide results related to the adequacy of the inservice test procedures and methods.

INSPECTABLE AREA: **Large Containment Isolation Valve Leak Rate and Status Verification**

Scope

Inspection activities in this area would be focused on the adequacy of the licensee's testing program for large containment isolation valves that provide a direct flow path from the containment atmosphere to outside containment. At most facilities the inspection scope would be limited to the containment purge and ventilation valves and personnel access hatches. Inspection activities related to leak rate testing for most of the containment isolation valves and/or containment Integrity issues would be captured by the corrective action program inspection activities.

Basis

Inspection in this area supports the Barrier Integrity cornerstone.

The inspection activities are intended to verify that the licensee has an acceptable process for insuring that major containment isolation valves will function as designed in preventing the release of contamination following a design basis accident.

The normal containment ventilation isolation valves tend to be very large valves with soft rubber seats. Industry experience has shown that the seats tend to dry out over long periods and fail to maintain their leakage Integrity. Pressurized water reactor containment purge valves are routinely opened during plant operation to purge the containment or to allow reductions in containment pressure. The constant cycling results in degradation of the valve seats. In both cases inspection efforts would be focused on insuring that the valves continue to meet the design leakage requirements and that the maintenance and testing efforts are appropriate.

Performance Indicators

There is a performance indicator for total leakage from all containment penetrations. However; the limited inspection activities detailed in this inspectable area will be used to verify the accuracy of the PI.

INSPECTABLE AREA: Licensed Operator Requalification

Scope

Inspection activities in this area would focus on the effectiveness of the licensee's program for conducting operator requalification training. Inspection activities would include a review of requalification examinations, administration of requalification examinations, the training feedback system and the remedial training program. In addition, inspection activities would verify that the facility's operating history has been factored into the requalification program and would verify conformance with operator license conditions.

Basis

Inspection of this area supports the Mitigating Systems, Barrier Integrity and Emergency Preparedness cornerstones because it can assess operator performance adequacy in responding to events.

This inspection evaluates operator performance in mitigating the consequences of events. Poor operator performance results in increased risk due to its impact on the human factors terms, assumed operator recovery rates and personnel induced common cause error rates assumed in the facility IPEs. Human performance errors and failure to recover from accident events are the most risk important events at a facility.

Performance Indicators

There are no performance indicators that have been established that can provide results related to the adequacy of the licensee's licensed operator requalification program.

INSPECTABLE AREA: **Maintenance Rule Implementation**

Scope

The inspection includes a review of goal setting, performance monitoring, repetitive failure determinations, and evaluations of functional failures and maintenance preventable functional failures. The scope of the inspection activities would include those systems covered under the maintenance rule which would also include a review of the licensee's implementation of the maintenance rule requirements for those systems.

Basis

Inspection of this item supports the Initiating Events, Mitigating Systems and Barrier Integrity cornerstones by assessing the effectiveness of the licensee program in ensuring availability and reliability of plant equipment.

Proper implementation of the maintenance rule is important to ensure reliable operation of plant equipment within the scope of the rule. The program should ensure that there is a proper balance that optimizes availability and reliability when removing equipment from service for preventive maintenance. High availability and reliability result in a high probability that accident mitigation systems will perform successfully when needed and that Barrier Integrity will remain effective in preventing the release of radioactivity.

Performance Indicators

This inspection area supplements the safety system performance indicator (system unavailability). In addition, inspection activities in this area would provide an assessment of equipment reliability where a performance indicator does not exist.

INSPECTABLE AREA: **Maintenance Work Prioritization and Control**

Scope

Inspection activities in this area would focus on the effectiveness of the licensee's programs for work prioritization and control during shutdown and power operations. Licensee work prioritization methodologies, level of maintenance support, and assessments of integrated risk of the work backlog would be reviewed by the inspector.

Basis

This inspection item supports the Mitigating Systems, Initiating Events and Barrier Integrity cornerstones.

Maintenance is the primary means of mitigating and managing the effects of component degradation and failures. Operating experience shows that the lack of maintenance (component deficiencies not corrected) or improperly performed maintenance (maintenance activities not well controlled) can greatly contribute to the risk for event initiation, and may cause SSCs to not function properly if called upon to mitigate the consequence of an event. Operating experience also shows that for risk significant events identified through the Accident Sequence Precursor (ASP) program, work control and failure to maintain equipment represent the majority of causes. Appropriate identification, prioritization, planning, scheduling, and completion of risk significant work is essential to safe operation.

One specific area that should be included in inspection of this area is the control of risk significant work in the switchyard. A large percentage of loss-of-offsite power events occurred when either some major electrical power source was out of service prior to the event and/or some major electrical power source failed during the event. It is important that work occurring in the switchyard be well controlled to prevent an unplanned loss of a power source due to maintenance errors. Also, the simultaneous removal of multiple electrical power sources from service should be avoided, particularly during shutdown conditions.

Performance Indicators

There are several performance indicators (PIs) that indirectly infer the quality of work prioritization and control to reduce inspection effort in this area. However, events that have reached the ASP threshold (E-06) tended to be random and were not predicted through existing PIs. This inspection supplements these PIs.

INSPECTABLE AREA: **Nonroutine Plant Evolutions**

Scope

The inspection activities will be used to evaluate operator and equipment performance for other than normal and routine operations. This inspection activity will provide a vital tie between operator performance observed under simulated conditions and those observed during non routine plant operations. This activity will also provide a snapshot of plant and equipment performance during transient conditions.

Basis

This inspection primarily supports the Mitigating Systems and Barrier Integrity cornerstones by providing assessment of operator performance during transient and off-normal operations. Poor operator performance could also affect the Initiating Events cornerstone. In addition to providing observations of non routine plant operations, inspections in this area provide increased opportunities to observe more significant plant transients and to evaluate operator and equipment performance during those non-simulated transient conditions.

Operator performance provides a vital link in mitigating the consequences of improper or unforseen equipment performance. Degrading operator performance results in increased risk due to its impact on human factors terms, assumed operator recovery rates, and personnel induced common cause errors. Probabilistic risk assessments have shown that human errors can be very significant contributors to risk, in particular during recovery from accident events.

Performance Indicators

Operator performance under abnormal plant operating conditions cannot be sufficiently covered by a performance indicator because the PIs of transients and trips do not include near-miss events

INSPECTABLE AREA: **Operability Evaluations**

Scope

Inspection activities in this area would focus on the effectiveness of the licensee's program for the evaluation of degraded and non-conforming conditions affecting plant systems, structures and components (SSCs). Inspection activities would be limited to a review of those potentially risk significant degraded and non-conforming conditions affecting SSCs that are considered to be operable and fully capable of performing their design functions based on written operability evaluations. Initial reviews of the operability evaluations should be performed following formal completion of the evaluations by the licensee.

Basis

Inspection of this item primarily supports the Mitigating Systems by ensuring risk-significant SSCs are fully functional to perform their design function.

The inspection activities are intended to verify that the licensee has taken the necessary steps to demonstrate that the reliability, availability and functional capability of the SSCs and associated components are maintained although the SSCs are degraded and/or non-conforming in some way.

As a result of the size and complexity of a nuclear power plant, degraded and non-conforming conditions are frequently identified at all plants. Risk-significant SSCs are often affected and the degraded or non-conforming condition cannot always be corrected immediately. An improperly evaluated degraded and/or non-conforming condition may result in continued operation with a SSC that is not capable of performing its design function which would result in operation of the plant outside of its design and license bases. The potential effects on safe operation could include the loss of redundancy within a safety system, the loss of safety function or a reduction in the safety margin assumed in the plant design and analyses.

The inspection would ensure that the evaluations include an adequate technical justification to support the operability evaluation and would verify the implementation of any compensatory measures.

Performance Indicators

There are no performance indicators that provide effective assessment of the quality of operability evaluations.

INSPECTABLE AREA: Operator Work-Arounds

Scope

Inspection activities in this area would focus on plant and control room deficiencies that have the potential to affect performance in conducting routine and non-routine evolutions. Detailed inspection activities would be limited to those risk significant deficiencies that could compromise equipment and personnel mitigation strategies. The inspection would focus on those deficiencies that are not included in the temporary modification process and would require operator actions that are in addition to those assumed in the initial design.

Basis

Inspection of this area supports the Mitigating Systems cornerstone.

Operator work-arounds can have an adverse effect on the functional capability of a system in that the system may not be capable of performing its design function without operator intervention. An excessive number of operator work-arounds or those requiring complex operator actions reduce the effectiveness of the operations staff in responding to transient conditions and will increase the chance of operator errors. PRAs have identified human errors as significant contributors to risk.

Performance Indicators

There are no performance indicators that have been established that can provide results related to the adequacy of the licensee's process for controlling operator work arounds. Performance indicators can not assess the significance of operator work-around items.

INSPECTABLE AREA: Permanent Plant Modifications

Scope

Inspection activities in this area include the review of design, installation, configuration control, and post-modification testing for the potentially risk significant permanent modifications of the SSCs covered by the maintenance rule. Inspection activities would also include an in-depth review of changes to the initial licensed design, design basis documents, test procedures and normal and emergency operating procedures.

Inspection Basis

Inspection of this area supports the Mitigating Systems and Barrier Integrity cornerstones.

Inspection of permanent plant modifications provides monitoring of the licensee's performance to ensure that the design bases for risk-significant systems, structures, and components (SSCs) have been maintained and that the changes have not adversely affected the licensing and design bases and safety functions of the SSCs. Plant modifications may introduce changes to the assumptions and models used in the plant specific PRA. Modifications to one system may affect the design bases and functioning of other interfacing systems. Also, similar modifications to several systems could introduce potential for common cause failures that affect plant risk.

Industry experience has shown that modifications to risk-significant SSCs can adversely affect their availability, reliability or functional capability. The baseline Inspection of permanent modifications should focus on: (1) compliance with regulations, (2) consistency with defense-in-depth philosophy, (3) maintaining sufficient safety margins, and (4) acceptability of the effects on risk.

Verification of post-modification testing to confirm that the objectives of the modification are met and verification that the system is restored to the required configuration after completion of the modification are important. Design requirements that cannot be verified by testing of the modification, such as seismic or environmental qualifications should also be reviewed.

Performance Indicators

No performance indicators have been established that can provide results related to the adequacy of permanent modifications.

INSPECTABLE AREA: **Physical Protection System (Barriers, Intrusion Detection System, and Alarm Assessment)**

Scope

Verify that the licensee has an effective physical protection system in place capable of providing high assurance that the facility is protected against the external threat of radiological sabotage. The system includes protected and vital area barriers, associated intrusion detection systems, and alarm assessment capabilities.

Basis

Inspection of this area supports the Physical Security cornerstone.

This is a risk significant system that is necessary for protection against the external threat of radiological sabotage. Operability of the protected area intrusion detection system and of the vital area intrusion detection system is necessary to identify and initiate response to security events. The system is the first line of defense in the "defense in depth" concept of protection against radiological sabotage. The risk significance is based on an exploitable vulnerability by a person(s) with the intent and capability of committing radiological sabotage. The frequency of occurrence of this type event has been low. However, the consequences of such an event would be moderate to high.

Performance Indicators

This area will be assessed by a performance indicator after the initial verification and validation inspection is done to confirm implementation is acceptable and that reporting thresholds for significant event meet regulatory expectations. The initial verification and validation will serve to ensure valid data is used for the PIs.

The performance indicator for this area measures that each of these systems can perform their intended function 95% of the time. For example, if compensatory posting hours for the perimeter intrusion detection system exceeds approximately 438 hours in 12 months, a supplemental inspection may be performed to address this specific area. If the protected area or the vital area system falls below the 95% indicator, a supplemental inspection will be performed for the entire system. The percent of time equipment is available and capable of performing its intended function will provide data on the effectiveness of the maintenance process and provides a method of monitoring equipment degradation because of ageing that could adversely impact on reliability. The reporting of equipment percent availability will be accompanied by the reporting of compensatory hour for equipment out of service due to extreme environmental conditions (severe storms, heavy fog, heavy snowfall, sun glare that renders the assessment system temporarily inoperative, etc.,) and for planned maintenance and modifications. The extreme environmental and planned maintenance and modifications compensatory hours will not be considered as equipment unavailability as part of the PI but are part of the total compensatory hours and will provide information on events that are contributing to equipment unavailability. The data in this area will be reported as two PIs, one as percent availability for the protected area system and one for the vital area system. This indicator is considered adequate to assess performance and no additional inspection of this area is necessary.

INSPECTABLE AREA: Piping System Erosion/Corrosion

Scope

The inspection activities in this area would focus on the effectiveness of the licensee's erosion and corrosion program. Inspection activities would include reviews of the licensee's monitoring, detection and correction of piping and component degradation caused by erosion and/or corrosion. Inspection activities would ensure that SSCs were being adequately monitored and that appropriate corrective actions were implemented. Inspection activities would be limited to the site specific risk significant SSCs and would include reviews of system test results and reviews of corrective actions for identified deficiencies.

Basis

Inspection of this item supports the Initiating Events cornerstone.

The inspection activities are intended to verify that the licensee has adequately implemented the erosion/corrosion program so that the SSCs remain reliable and fully functional.

Effective implementation of an erosion/corrosion program is important to minimize the potential for high energy fluid system failures that can result in plant transients, damage to plant equipment and/or injury of personnel. The industry has experienced failures of steam system piping as a result of the effects of erosion/corrosion which underscore the importance of monitoring licensee performance in this area. Effective implementation of the erosion/corrosion program is also important to minimize the potential for internal flooding or a loss of system function.

Performance Indicators

There are no performance indicators established that can provide results related to the adequacy of the erosion/corrosion program before an SSC degrades or fails.

INSPECTABLE AREA: **Post Maintenance Testing**

Scope

Inspection activities would focus on verification that the post maintenance test procedures and test activities were adequate to verify system operability and functional capability for the maintenance that was performed. The inspection would focus on significant maintenance involving high risk significant systems or components, in areas that have the potential to cause common mode/cause failures, where repetitive failures indicate programmatic problems, or on maintenance activities that have the potential to significantly impact risk.

Basis

Inspection of this item primarily supports the Mitigating Systems cornerstone.

This is the only process available to verify that a system or component is reliable and fully functional following maintenance.

Post maintenance testing provides the final check that a system and /or component has been returned to its required design configuration and will perform its design function(s) following completion of maintenance activities. Inadequate maintenance activities that are not detected prior to returning the equipment to service can result in a significant increase in unidentified risk for the subject system and in common mode/cause failures and potential for loss of function on redundant trains and identical components in other systems.

Performance Indicators

This inspection activity will supplement PIs and maintenance rule implementation inspectable area. The PIs do not measure the adequacy of the post-maintenance test procedures.

INSPECTABLE AREAS: Radiation Monitoring Instrumentation

Scope

Inspection of this area should ensure that criticality, area radiation monitors (ARMs), continuous air (CAMs) and applicable Radiation Monitoring System (RMS) monitors are reliable and accurate in areas where activities could result in transient HRAs, VHRAs or airborne areas. This inspection will also include the containment dome monitors, because of their importance in accident analysis and classification, and portable instrumentation used to assess radiologically significant areas or activities (such as underwater meters used during diving). However, the inspection will not include those monitors that a licensee has included under their Maintenance Rule program.

Basis

Inspection in this area supports the plant facilities/equipment and instrumentation attribute of the Occupational Exposure cornerstone

This inspection will verify that these monitors are calibrated and maintained (including verifying alarm setpoints) as required by 10 CFR Part 20 or a licensee's technical specifications and procedures

Radiological risk (i.e., exposure) to a worker should be maintained within the occupational exposure limits defined in 10 CFR Part 20 and ALARA and to minimize the potential for health effects. These monitors identify changing radiological conditions to workers such that actions to prevent an overexposure can be taken. Industry has experienced several events where these monitors were the primary indication that radiological conditions had significantly changed as a result of planned or unplanned activities.

Performance Indicators

None of the established PIs cover this area. Monitor locations are site specific and assessments of their reliability and accuracy will require baseline inspection.

INSPECTABLE AREA: Radiation Worker Performance

Scope

Inspection activities in this area consists of observing radiation worker (including Radiation Protection and Chemistry (RP&C) technicians) performance to verify that they aware of and use appropriate radiological controls (such as properly controlling radioactive material) when performing work in radiological areas.

The focus is on whether licensee identified radiation worker performance events were appropriately corrected, were not recurrent and were being trended to identify underlying performance issues (such as poor training). This includes observations of work during plant walk downs, performed as part of other inspectable areas (such as ALARA Planning and Controls and Radioactive Material Processing and Shipping), to verify that workers (including technicians) understand and use appropriate controls to maintain exposures within regulatory limits and ALARA, and to prevent an unauthorized release of radioactive material to the environment.

Basis

Inspection in this area supports the Occupational and Public Exposure cornerstones.

The objective of this area is to verify that workers understand the radiological hazards associated with nuclear plant operation, effectively identify and control these hazards, identify and resolve adverse trends or deficiencies, and maintain proper oversight of work.

The associated risk is the potential for a significant, unplanned exposure resulting either directly or in part by the failure of a worker to perform a required task owing to poor knowledge or training. Recurrent problems in this area have been identified by the industry as a root or contributing cause in many exposure events and in some events involving the unplanned release of radioactive material to the environment. This is of special concern during outages, when radiologically significant work is often performed by contract staff having varying levels of experience.

Performance Indicators

None of the established PIs cover this area.

INSPECTABLE AREA: **Radioactive Material Processing and Shipping**

Scope

Inspection of this area should verify that appropriate controls are instituted for the processing and shipping of radioactive material to a burial site or other licensed recipient. The inspection focus is to review the administrative and physical controls for radiologically significant activities (Type A, Type B, and higher risk material shipments) that prevent an inadvertent exposure to workers and the public. This includes observing selected shipping activities having some risk-significance (such as Type B or irradiated fuel shipments), including reviewing associated shipping records, to provide independent validation of the shipping program. Particular emphasis should be given to the 10 CFR Part 61 waste characterization and stability requirements (III.A.3 and III.C.5 of Subpart G to 10 CFR Part 20) as industry experience has shown this area has not been well addressed in licensee assessments. However, the inspection should use licensee assessments to review minor shipping activities, administrative controls, worker training and qualifications and to verify that significant changes to the DOT or NRC transportation requirements were addressed. All transportation events reported to the Commission or to the licensee should also be reviewed.

Basis

This inspection supports the program/process attribute of the Public Exposure cornerstone.

This inspection will verify that the radioactive material processing and shipping program complies with the requirements of 10 CFR Parts 20 and 71 and DOT regulations 49 CFR Parts 170-189. Radioactive material intended for burial must also comply with 10 CFR 61.55 - 61.57 waste classification and stability requirements.

The regulations state specific physical and administrative controls that provide for a layered defense against unplanned radiation exposure during radioactive material processing and transport or from an accidental breech of the shipping container. Although there is a low frequency of industry events, the actual or potential consequence (i.e., significant exposures or release of radioactive material) is typically high. Additionally, the NRC has determined that an independent assessment of performance in this area is necessary to ensure that adequate protection of public health and safety is maintained.

Performance Indicators

There is no established Performance Indicator for this area given the low frequency of events.

INSPECTABLE AREA: Radiological Environmental Monitoring Program (REMP)

Scope

Inspection of this area will ensure that the REMP reasonably measures the effects of radioactive releases to the environment and sufficiently validates the Integrity of the gaseous and liquid effluent release program. The focus is on adverse trends or recurrent problems identified through licensee assessments or the Annual Environmental Monitoring Report and periodic observations of worker and equipment performance. The inspection should not focus on the quality of procedures or minor administrative processes as these will be addressed in the licensee assessments.

The baseline inspection should consist of an in-office review of the Radiological Environmental Monitoring Report and routine licensee assessment results to verify that the REMP was implemented as required by the Technical Specifications and the ODCM. Specific emphasis should be placed on verifying that environmental sampling is representative of the release pathways and that missed samples and/or inoperable sampling/analyses equipment are being properly addressed. A subsequent on-site walkdown to observe sampler stations, environmental sampling and analyses techniques, and to review the calibration and maintenance of the counting room instrumentation should also be performed. The inspection should not focus on the quality of procedures, tracking of samples or other minor administrative processes as deficiencies in this area would normally be identified through the licensee assessments.

Basis

Inspection in this area supports the plant facilities/equipment and instrumentation, and program/process attributes of the Public Exposure cornerstone.

This inspection will verify that the REMP is implemented consistent with the licensee’s technical specifications to validate that the effluent release program meets the ALARA principle of Section IV.B of Appendix I to 10 CFR Part 50.

The REMP supplements the effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are as predicted by the effluent measurements and modeling of effluent pathways. As such, it serves as the final Barrier Integrity in assuring that the associated dose from radioactive releases is within regulatory limits. Industry experience has shown that the REMP is often the primary method of assessing the potential risk from unplanned or unmonitored radioactive releases. Because REMP results have served to allay public concerns regarding the actual health effects due to radioactive releases associated with power plant operation, the NRC has determined that an independent assessment of performance in this area is necessary to ensure that adequate protection of the public health and safety is maintained.

Performance Indicators

None of the established PIs cover this area.

INSPECTABLE AREA: **Refueling and Outage Related Activities**

Scope

Inspection activities in this area would focus on the licensee's shutdown risk management program and those outage related activities that have the potential to impact the risk to the plant. These areas include plant cool down, transfer to shutdown cooling, solid operations, drain down evolutions, fuel handling (core off-load/reload), mid-loop/reduced inventory operations, containment integrity, plant heat up, reactor startup and physics testing. In addition, the inspection activities would include support systems necessary to mitigate the consequences of shutdown accidents, which includes control of switch yard activities, emergency diesel generator availability and vital power availability. Inspection activities in this area would include activities during forced or planned outages and would not be limited to only refueling outages.

Basis

Inspection of this item supports the Initiating Events, Mitigating Systems and Barrier Integrity cornerstones.

The inspection activities are intended to verify that the licensee has taken the necessary steps to minimize potential events, maintain defense in depth, ensure the appropriate SSCs are maintained available to mitigate and contain postulated accidents.

Due to changing plant configuration, combinations of equipment outages can place the plant in a condition where single failures can quickly lead to significant adverse conditions such as core boiling. In addition, operations and maintenance personnel are performing non-routine tasks which have greater risk impact due to the extensive amount of equipment that is usually out of service. These items, along with the fact that the barriers to prevent a radiological release are also degraded, result in a significant increase in risk if not appropriately controlled by the licensee.

Performance Indicators

There are currently no performance indicators that have been established that can provide results related to the licensee's performance during refueling outages. A shutdown performance indicator is under development.

INSPECTABLE AREA: **Response to Contingency Events (Protective Strategy and implementation of Protective Strategy)**

Scope

Verify that the licensee has the capability to protect its vital area target sets against the design basis threat. The implementation of the protective strategy includes demonstrating that the strategy works, and that security force can successfully protect against the design basis threat through drills and exercises.

Basis

Inspection in this area supports the Physical Security cornerstone.

This is a high risk-significant system necessary to protect against the design basis threat of radiological sabotage. The licensee should be able to demonstrate the ability to respond with sufficient force, properly armed, appropriately trained and within the appropriate time frame to protected positions in order to interdict and defeat the design-basis adversary force in order to protect vital equipment necessary for the safe shutdown of the plant.

The ability of the security force to effectively respond to the design basis threat is contingent upon the number of armed responders committed to in the physical security plan; the intrusion detection system being able to detect; the alarm status being communicated to the alarm stations; the assessment functions (closed-circuit television and lighting) and the training of CAS and SAS operators, communications on and off site, the response officers and response team leaders, including handling and qualification with assigned weapons, and the use of proper tactics. Each of these items will be reviewed to determine if they can perform their intended function in support of the design basis threat and as verification of the PI identified in the Physical Protection System inspectable area.

The consequence to radiological sabotage if an attack did occur is high.

Performance Indicators

None of the established PIs cover this area.

INSPECTABLE AREA: Safety System Design and Performance Capability

Scope

Inspection includes review of design bases, final safety analysis report (FSAR), supporting calculations, as-built conditions, modifications, testing, and normal and emergency operations of risk-significant systems and interfaces with support systems. This would be an in-depth review of a selected risk significant system and support systems with an emphasis on changes to the design bases and normal and emergency plant procedures.

Basis

Inspection of this area supports the Mitigating Systems cornerstone.

Inspection of safety system design and performance verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected system to perform its design basis functions. The inspection should focus on the design and functional capability of components that are not validated by in-plant testing. Also, seismic and environmental qualifications of the SSCs should be verified. The PRA assumptions and models are based on the ability of the as-built safety system to perform its intended safety function successfully. If the design bases of the system had not been correctly implemented in the installed system, the operation and test procedures, and the supporting analyses and calculations, the system cannot be relied upon to meet its design bases and performance requirements. The design interfaces with support systems, such as cooling systems, ventilation systems, and instrument air system, should also be reviewed.

The baseline inspection should focus on: (1) maintaining design bases (2) consistency with defense-in-depth philosophy, and (3) maintaining sufficient safety margins.

Performance Indicators

There are no performance indicators that have been established that can provide results related to correct implementation of the design bases in the as-built system and the associated plant documents.

INSPECTABLE AREA: Surveillance Testing

Scope

Inspection activities in this area would be focused on ensuring test procedures are adequate to confirm SSCs will perform in accordance with their design. The inspector would review test results for adequacy in meeting the requirements, observe ongoing testing to evaluate human performance, and ensure that appropriate test acceptance criteria is in agreement with design requirements.

Basis

Inspection of this area ensures that safety systems are capable of performing their safety function and would support the Mitigating Systems and Barrier Integrity cornerstones.

Surveillance activities are required to verify that systems and components are reliable and functionally capable of performing their design function. Surveillance testing is the minimum required testing specified in the facility license and ensures that a conservative safety margin exists for system capability. Operating experience has shown that test procedure deficiencies may invalidate previously acceptable test results.

Performance Indicators

The PIs indirectly verify the adequacy of required surveillance test activities. The inspection is performed to supplement the PIs.

INSPECTABLE AREA: Temporary Plant Modifications

Scope

Inspection activities in this area includes a review of design, installation, configuration control, and post-modification testing for potentially risk significant temporary modifications of the SSCs covered by the maintenance rule.

Basis

Inspection of this area supports the design and design control attributes of the Mitigating Systems and Barrier Integrity cornerstones.

Inspection of temporary plant modifications provides monitoring of the licensee's performance in ensuring that the design bases for risk-significant systems, structures, and components (SSCs) have been maintained and that the changes have not adversely affected the safety functions of the SSCs. Temporary modifications may introduce change to the assumptions and models used in the plant specific PRA. A temporary change to one system may affect the design bases and safety functions of other interfacing safety systems. An increase in the likelihood of the occurrence of an initiating event could result from a temporary change. Also, similar temporary modifications to several systems could introduce the potential for common cause failures that affect plant risk.

Industry experience has shown that temporary modifications to risk-significant SSCs can adversely affect their availability, reliability or functional capability. Verification that all safety functions of the system are restored after completion of the temporary modification is important.

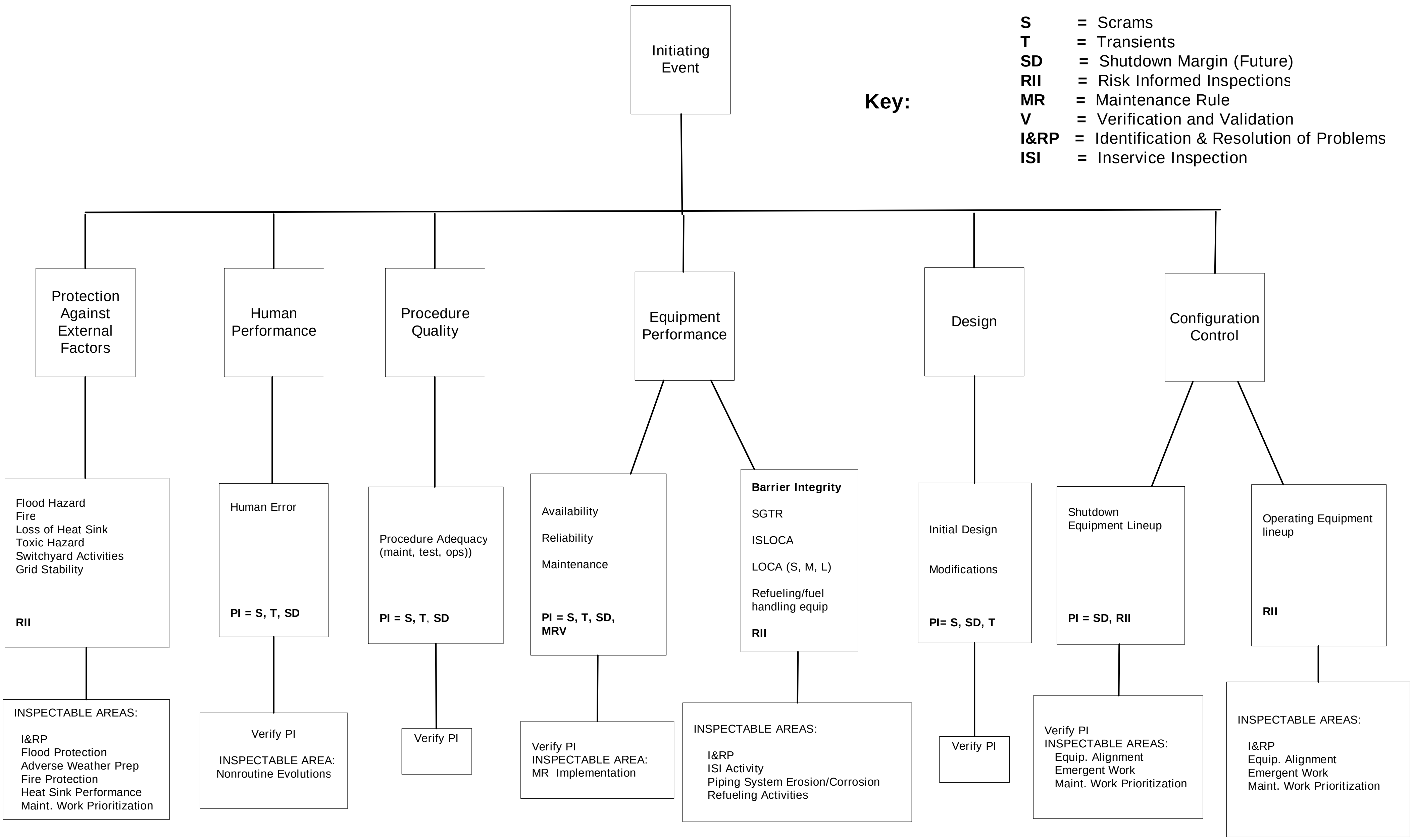
Performance Indicators

No performance indicators have been established that can provide results related to the adequacy of temporary modifications.

APPENDIX II

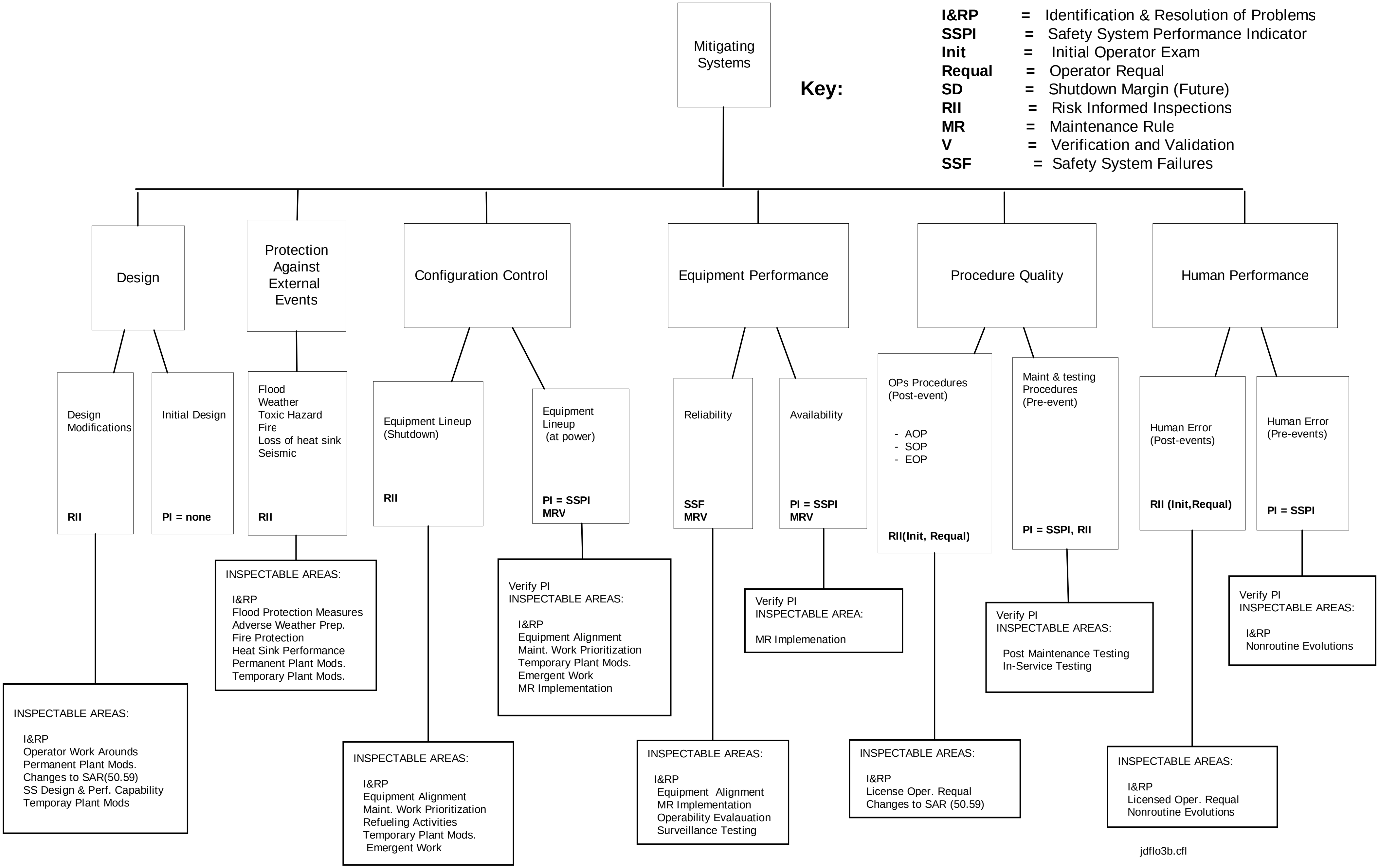
CORNERSTONE CHARTS

December 1998



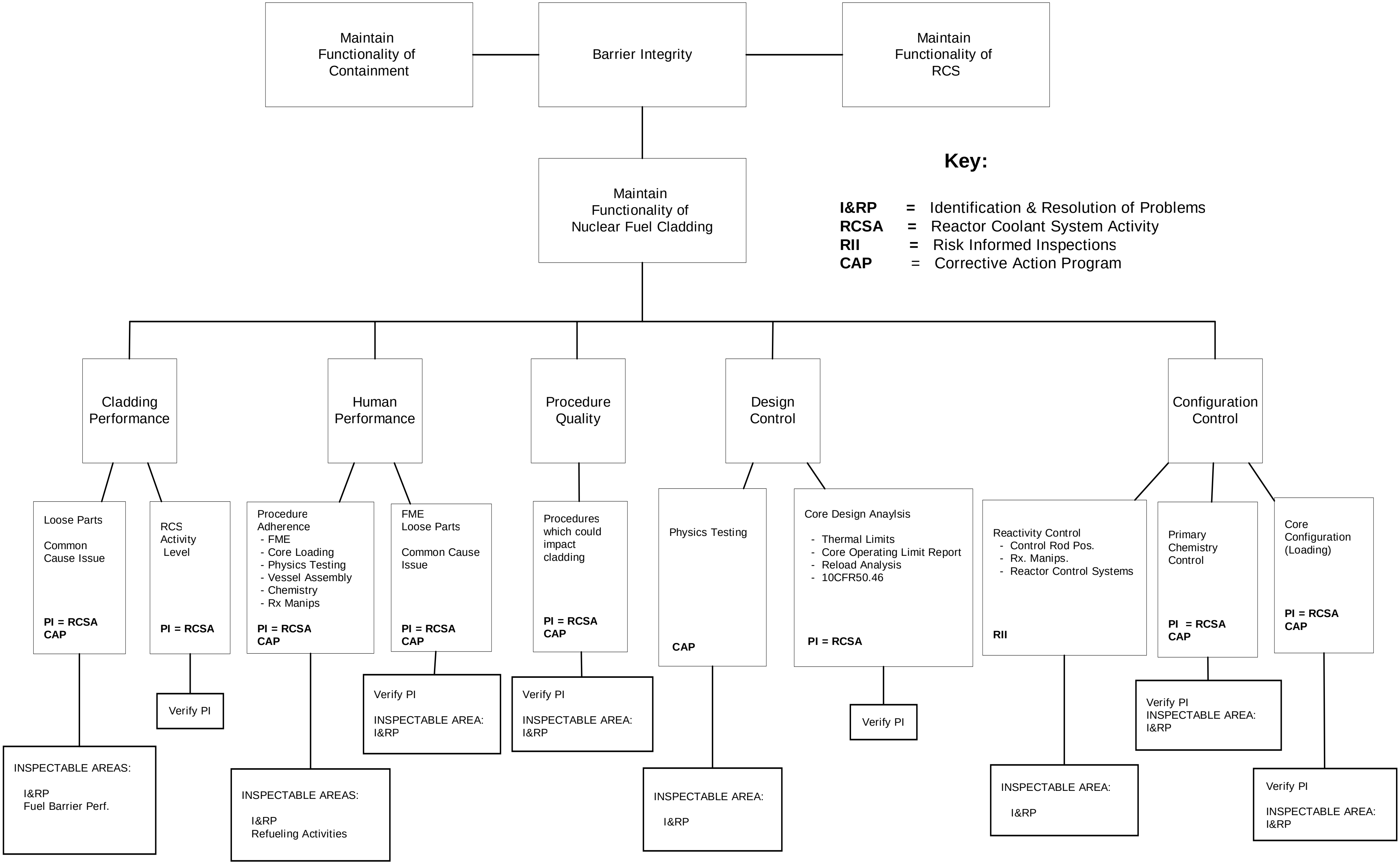
December 10, 1998

jdflo1B.cfl

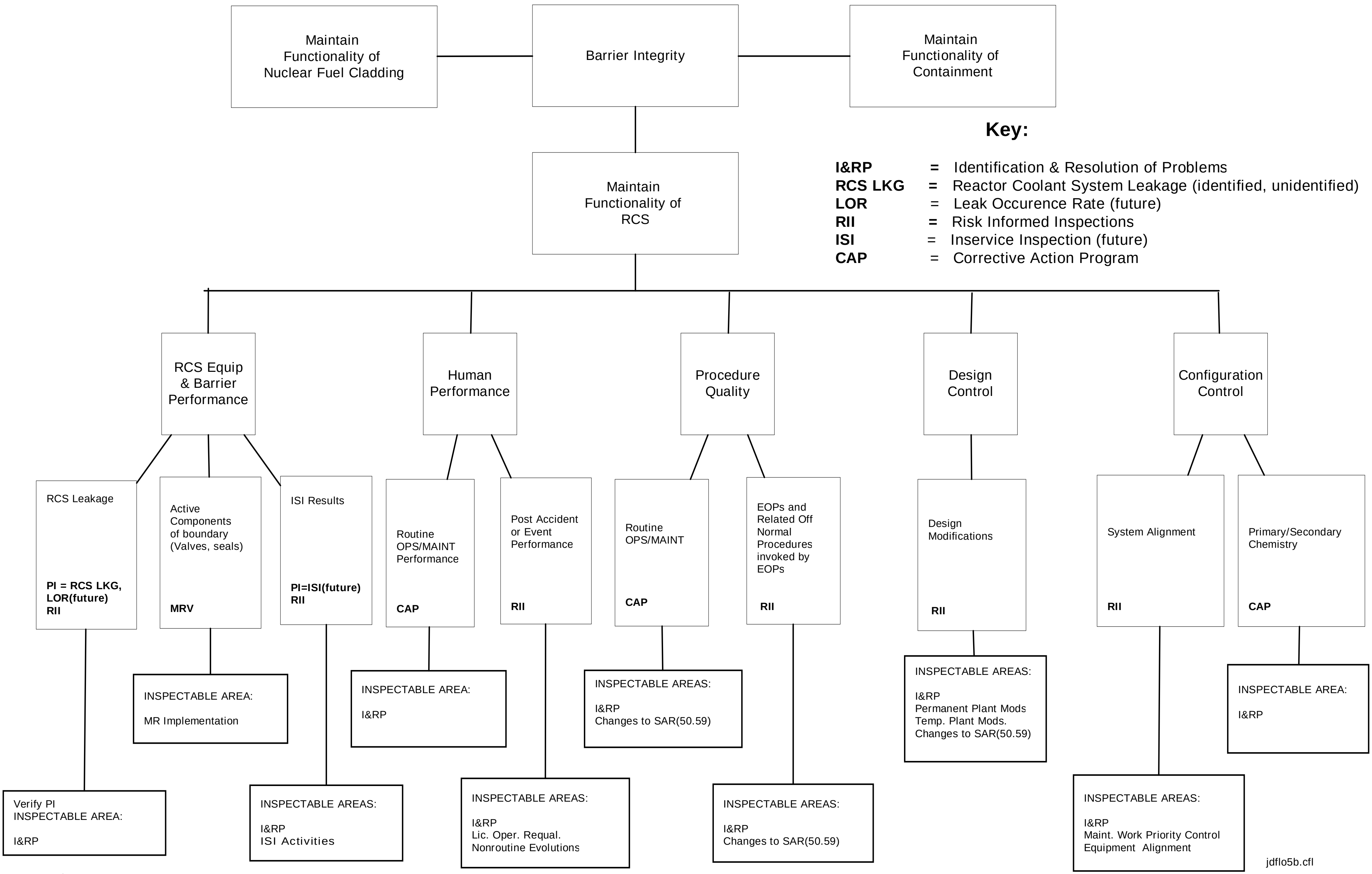


jdflo3b.cfl

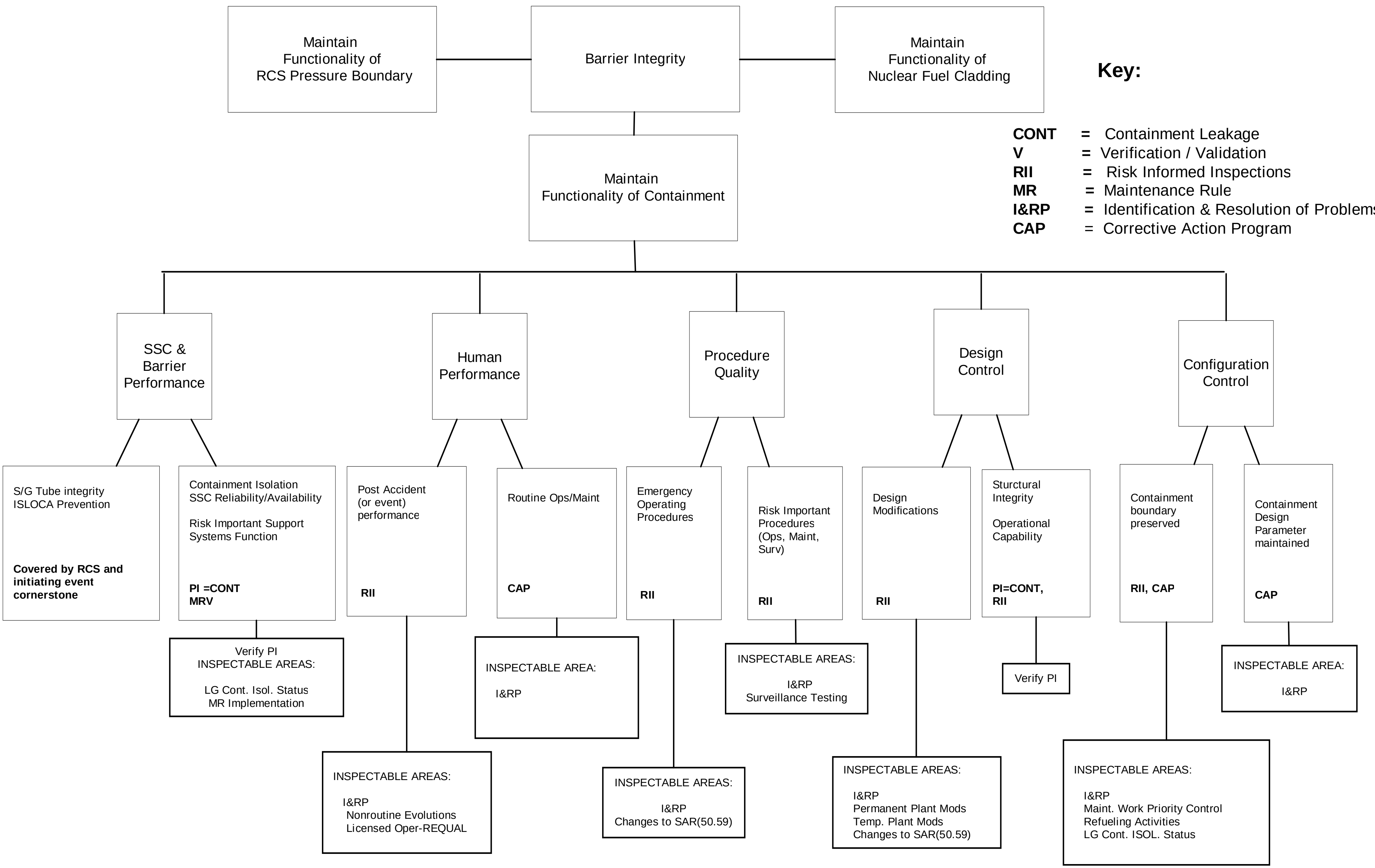
December 11, 1998


Maintain Functionality of Containment
Barrier Integrity
Maintain Functionality of RCS
Maintain Functionality of Nuclear Fuel Cladding
Key:
I&RP = Identification & Resolution of Problems
RCSA = Reactor Coolant System Activity
RII = Risk Informed Inspections
CAP = Corrective Action Program
Cladding Performance
Human Performance
Procedure Quality
Design Control
Configuration Control
Loose Parts
Common Cause Issue
PI = RCSA
CAP
INSPECTABLE AREAS:
I&RP
Fuel Barrier Perf.
RCS Activity Level
PI = RCSA
Verify PI
Procedure Adherence
- FME
- Core Loading
- Physics Testing
- Vessel Assembly
- Chemistry
- Rx Manips
PI = RCSA
CAP
INSPECTABLE AREAS:
I&RP
Refueling Activities
FME
Loose Parts
Common Cause Issue
PI = RCSA
CAP
Verify PI
INSPECTABLE AREA:
I&RP
Procedures which could impact cladding
PI = RCSA
CAP
Verify PI
INSPECTABLE AREA:
I&RP
Physics Testing
CAP
INSPECTABLE AREA:
I&RP
Core Design Anaylsis
- Thermal Limits
- Core Operating Limit Report
- Reload Analysis
- 10CFR50.46
PI = RCSA
Verify PI
Reactivity Control
- Control Rod Pos.
- Rx. Manips.
- Reactor Control Systems
RII
INSPECTABLE AREA:
I&RP
Primary Chemistry Control
PI = RCSA
CAP
Verify PI
INSPECTABLE AREA:
I&RP
Core Configuration (Loading)
PI = RCSA
CAP
Verify PI
INSPECTABLE AREA:
I&RP

jdflo4b.cfl

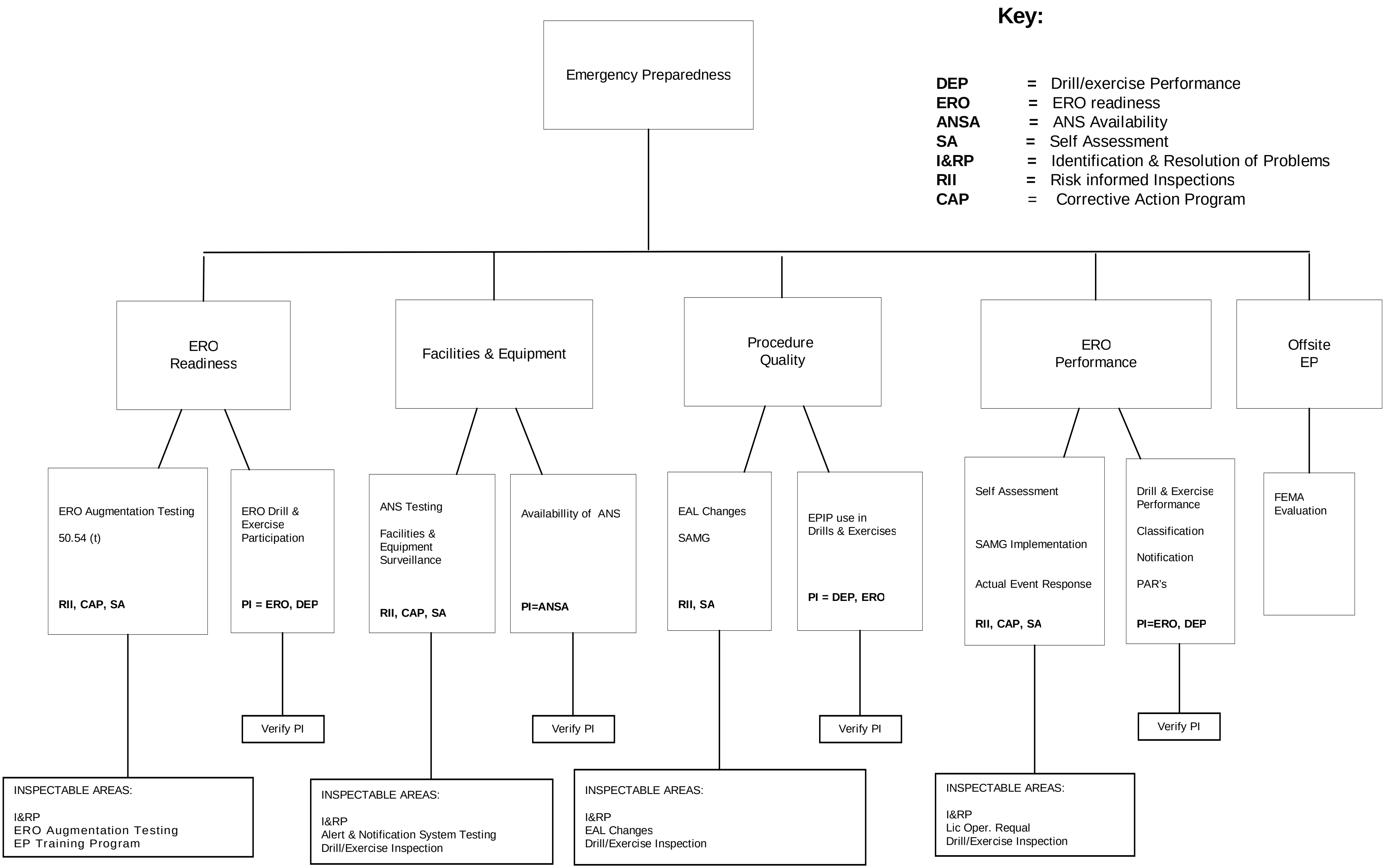


December 11, 1998



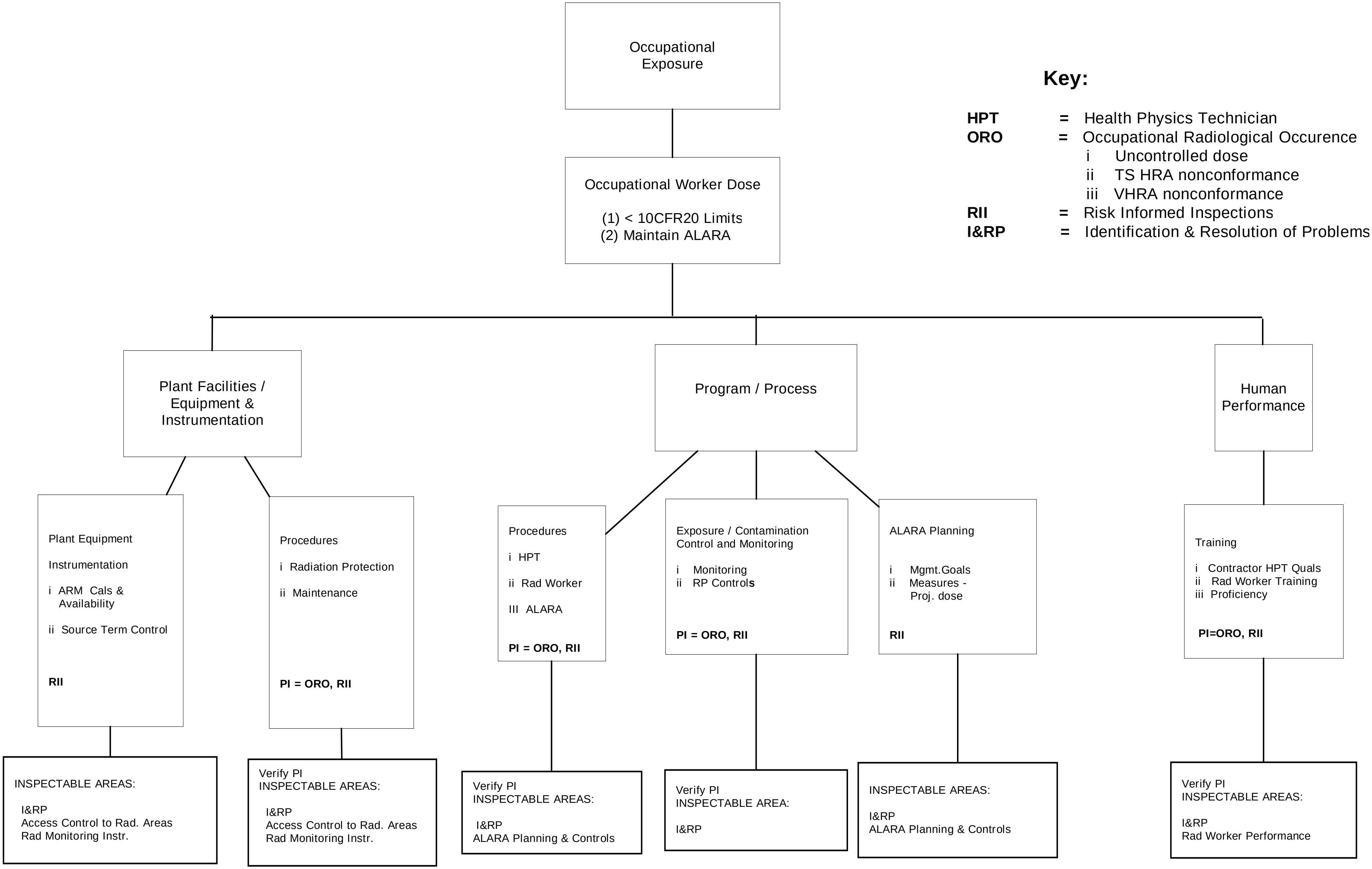
December 10, 1998

jdflo2B.cfl



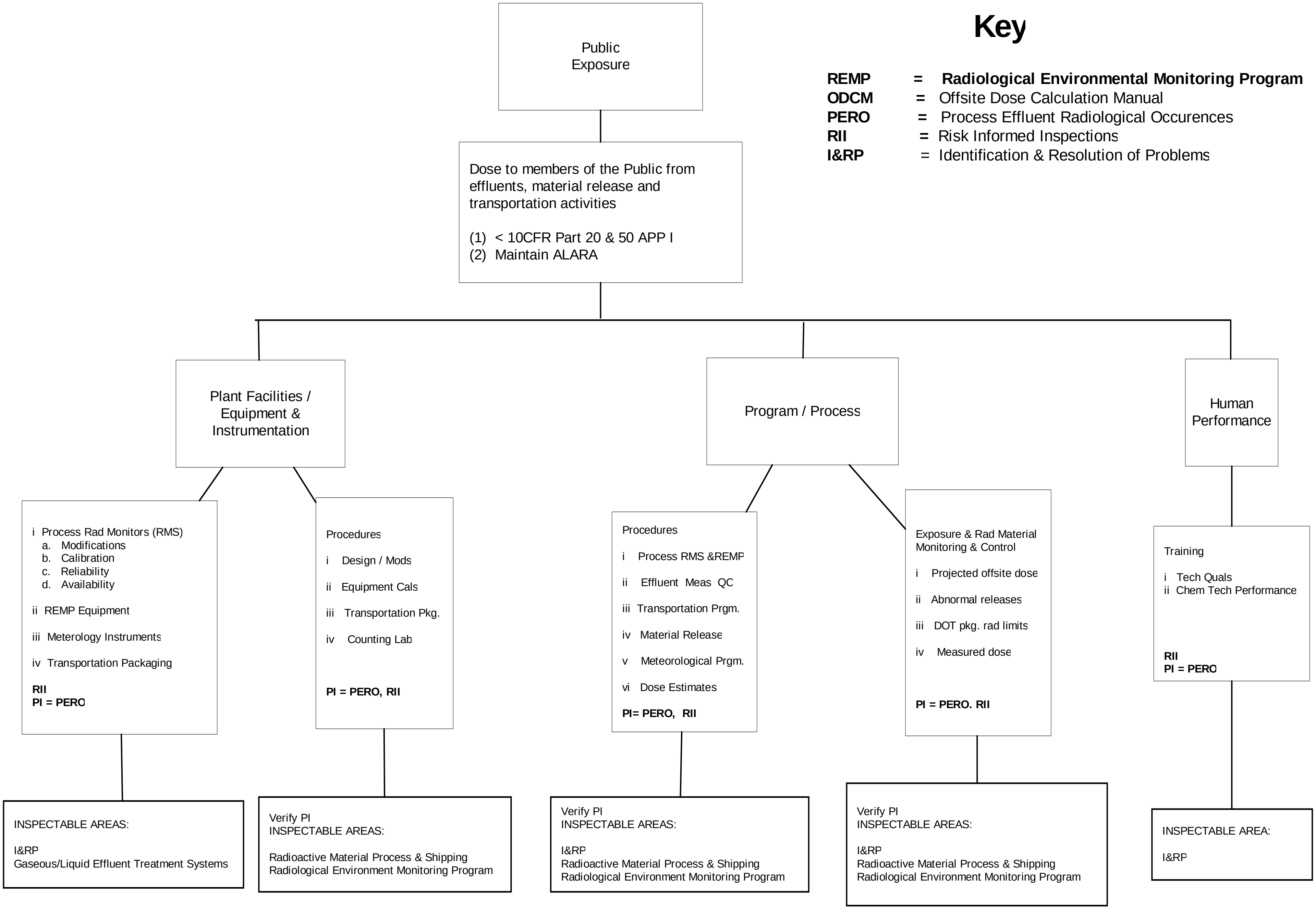
December 10, 1998

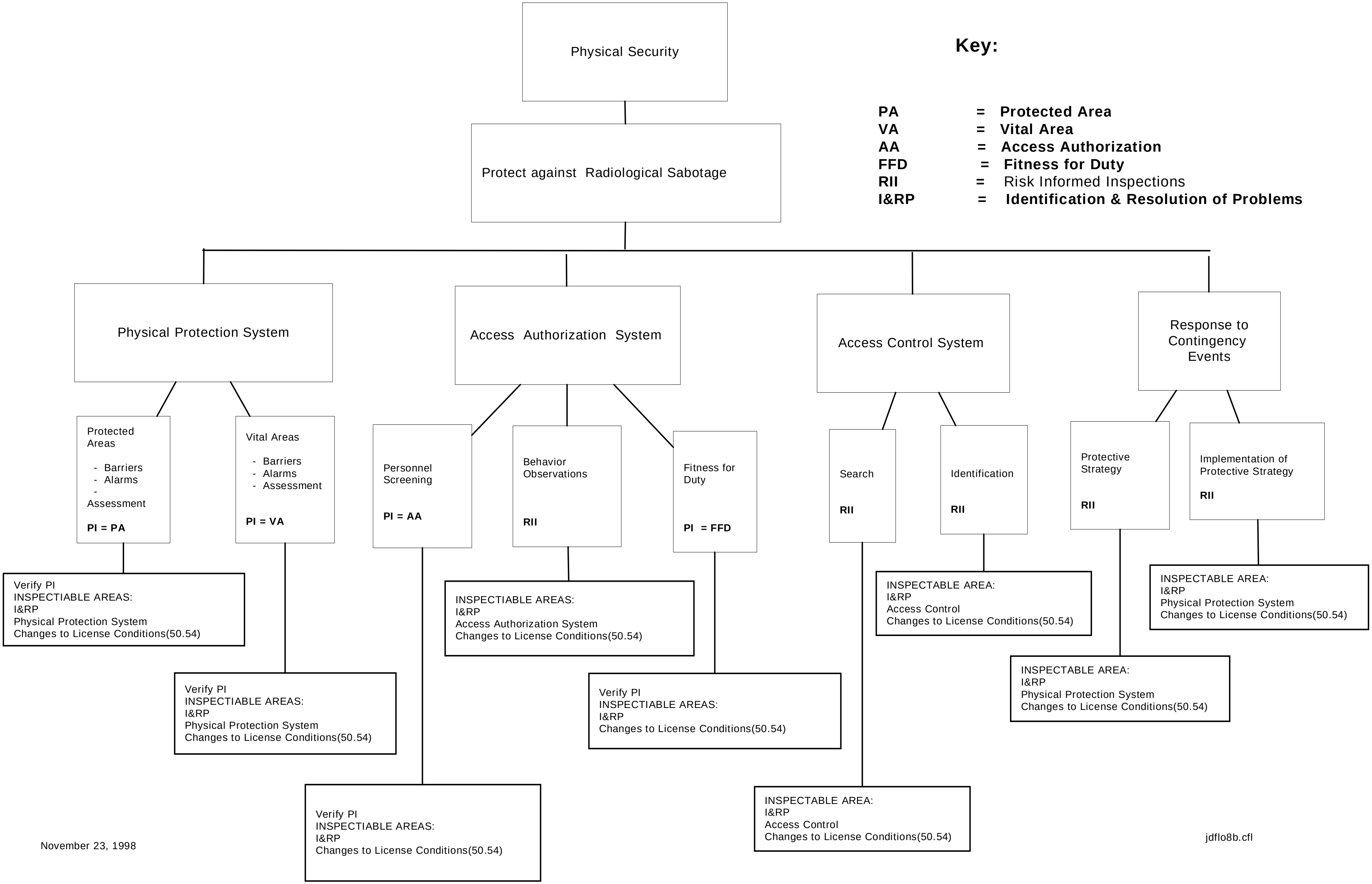
jdflow9b.cfl



December 10, 1998

jdflo6b.cfl


Key
REMP = Radiological Environmental Monitoring Program
ODCM = Offsite Dose Calculation Manual
PERO = Process Effluent Radiological Occurences
RII = Risk Informed Inspections
I&RP = Identification & Resolution of Problems
Public Exposure
Dose to members of the Public from effluents, material release and transportation activities
(1) < 10CFR Part 20 & 50 APP I
(2) Maintain ALARA
Plant Facilities / Equipment & Instrumentation
Program / Process
Human Performance
i Process Rad Monitors (RMS)
a. Modifications
b. Calibration
c. Reliability
d. Availability
ii REMP Equipment
iii Meterology Instruments
iv Transportation Packaging
RII
PI = PERO
Procedures
i Design / Mods
ii Equipment Cals
iii Transportation Pkg.
iv Counting Lab
PI = PERO, RII
Procedures
i Process RMS &REMP
ii Effluent Meas QC
iii Transportation Prgm.
iv Material Release
v Meteorological Prgm.
vi Dose Estimates
PI= PERO, RII
Exposure & Rad Material Monitoring & Control
i Projected offsite dose
ii Abnormal releases
iii DOT pkg. rad limits
iv Measured dose
PI = PERO. RII
Training
i Tech Quals
ii Chem Tech Performance
RII
PI = PERO
INSPECTABLE AREAS:
I&RP
Gaseous/Liquid Effluent Treatment Systems
Verify PI
INSPECTABLE AREAS:
Radioactive Material Process & Shipping
Radiological Environment Monitoring Program
Verify PI
INSPECTABLE AREAS:
I&RP
Radioactive Material Process & Shipping
Radiological Environment Monitoring Program
Verify PI
INSPECTABLE AREAS:
I&RP
Radioactive Material Process & Shipping
Radiological Environment Monitoring Program
INSPECTABLE AREA:
I&RP


Physical Security
Key:
PA = Protected Area
VA = Vital Area
AA = Access Authorization
FFD = Fitness for Duty
RII = Risk Informed Inspections
I&RP = Identification & Resolution of Problems
Protect against Radiological Sabotage
Physical Protection System
Access Authorization System
Access Control System
Response to Contingency Events
Protected Areas - Barriers - Alarms - Assessment PI = PA
Vital Areas - Barriers - Alarms - Assessment PI = VA
Personnel Screening PI = AA
Behavior Observations RII
Fitness for Duty PI = FFD
Search RII
Identification RII
Protective Strategy RII
Implementation of Protective Strategy RII
Verify PI INSPECTIABLE AREAS: I&RP Physical Protection System Changes to License Conditions(50.54)
Verify PI INSPECTIABLE AREAS: I&RP Physical Protection System Changes to License Conditions(50.54)
Verify PI INSPECTIABLE AREAS: I&RP Changes to License Conditions(50.54)
INSPECTIABLE AREAS: I&RP Access Authorization System Changes to License Conditions(50.54)
Verify PI INSPECTIABLE AREAS: I&RP Changes to License Conditions(50.54)
INSPECTABLE AREA: I&RP Access Control Changes to License Conditions(50.54)
INSPECTABLE AREA: I&RP Access Control Changes to License Conditions(50.54)
INSPECTABLE AREA: I&RP Physical Protection System Changes to License Conditions(50.54)
INSPECTABLE AREA: I&RP Physical Protection System Changes to License Conditions(50.54)

APPENDIX III

RISK INFORMATION MATRICES

December 1998

Contents

Table 1, Risk Information Matrix No. 1

Cornerstone	Inspectable Area	Performance Indicator	Frequency	Hours For 2-unit Site per Year	Level of Effort	Bases
EMERGENCY PREPAREDNESS (EP) CORNERSTONE						
EP	Alert and Notification System Availability	Alert and Notification System Availability	Biennial	4 hrs/2 yrs	Review all system and program changes. Expert judgement basis for hours estimation. After initial program verification (8 hours), Identification and Resolution of Problems/Issues review, and review system and program changes every 2 years. Total inspection hours required were reduced by 12 hours/2 years because of the availability of a PI.	Potential public exposure could be impacted by degradation in the ANS during an emergency.
EP	Drill and Exercise Inspection	Drill and Exercise Performance	Biennial Quarterly	64 hrs/2 yrs for specialist 14 hrs/yr for Resident or OL inspector	Observe the Biennial Exercise Expert judgement basis for hours estimation. PI in this area eliminates the need for additional inspectors to monitor the exercise (about 96 additional hours). Hours include observation of Biennial exercise by EP Specialist (32 hours every 2 years for observation, and 32 hours for Identification and Resolution of Problems/Issues). Resident Inspectors periodic observation of evaluation of Operator performance with respect to Emergency Plan during simulator observation (8 hours/year; one 2 hour observation per quarter). In addition, Resident should observe each annual exercise (6 hrs/yr).	Adequate performance between the licensee and the external agencies that respond to emergencies is necessary to limit potential public exposure.
EP	Emergency Action Level Changes	None	As required by program changes	16 hrs/2 yrs	Review all program changes. Expert judgement basis for hours estimation. Assumes modest EAL program change every 2 yrs. Extensive plan changes may require higher level of effort.	Timing of offsite response to emergencies can be impacted by inappropriate EAL changes, and this could impact potential public exposure.
EP	Emergency Plan Training	Drill and Exercise Performance, ERO Participation	Biennial	None	This inspection area is adequately covered by the PIs.	This inspection area is adequately covered by the Pis. Inadequate training could result in an increased potential for public exposure in the event of an accident.
EP	Emergency Response Organization Augmentation	None	Biennial	6 hrs/2 yrs	Review all licensee performed self assessments. Expert judgement basis for hours. Identification and Resolution of Problems/Issues review based on licensee self assessments.	Inadequate staffing during an emergency could impact licensee response, and could impact potential public exposure.
EP	Identification and Resolution of Problems/Issues			Incorporated in each applicable inspectable area		

Table 1, Risk Information Matrix No. 1 (continued)

CORNERSTONE	INSPECTABLE AREA	PERFORMANCE INDICATOR	FREQUENCY	HOURS FOR 2-UNIT SITE PER YEAR	LEVEL OF EFFORT	BASES
OCCUPATIONAL EXPOSURE (OE) CORNERSTONE						
OE	Access Control to Radiologically Significant Areas	High Radiation Area (for >1000 mR/hr only) Events, Very High Radiation Events, Significant Exposure Events	Annual	13 hrs/yr	Walkdown all HRAs < 1000 mrem/hr and select areas that are subject to transient dose rates, if those conditions exist Assure that proper controls have been established, and that workers understand access controls for these areas. Review those HRA access control events documented in the PIs and/or in the licensee's corrective action system over the last 6-12 months. The hours are based on the current core program as modified by inspection experience. The total hours include 9 hours of Identification and Resolution of Problems/Issues for the Region Based and the Resident Inspectors to review incidents involving the loss of one or more barriers to an HRA, VHRA or airborne area; Four hours of walkdown are included to observe radiologically significant work not addressed by the PI and to verify that HRAs $\leq$ 1000 mrem/hr are controlled as required by the applicable TS. Total inspection hours required were reduced because of the availability of a PI.	The potential for high occupational doses are higher in HRAs and in areas where transient HRA could exist. Radiological risk (i.e., exposure) to a worker must be within the occupational exposure limits defined in 10 CFR Part 20 and ALARA to minimize the potential for health effects. Collectively, the access controls provide a "defense-in-depth" against a significant exposure. Industry experience has identified frequent occurrences where the failure of multiple barriers resulted in an uncontrolled entry and, in some cases, a significant exposure
OE	ALARA Planning and Controls	None	Annual	60 hrs/yr	The inspectors shall review, at a minimum, the top five radiologically significant jobs. Select jobs having a high individual or collective dose or located in an HRA, VHRA, or airborne area by attending licensee planning and RP briefings and reviewing RP&C logbook entries and past outage histories. Compare current licensee performance to established exposure goals and previous performance, assess whether these goals were aggressive and reasonable, identify what exposure controls were implemented, and determine if the licensee's subsequent performance met these goals. Observe selected jobs to determine if the work is being performed as planned. The hours are based on the current core program as modified by inspection experience. The total hours include 10 hours of Identification and Resolution of Problems/Issues to review licensee assessments of the ALARA program and applicable events; 30 hours to review the planning for selected radiologically significant jobs; and 20 hours to observe those activities selected. It is expected that this inspection will be performed prior to (i.e., observe planning) and during (i.e., observe implementation) an outage. However, if no outage is scheduled for that site for the year, then this effort should only require about 40 total hours (i.e., 20 hours for job review and 10 hours for walkdown).	Radiological risk (i.e., exposure) to a worker be within the occupational exposure limits defined in 10 CFR Part 20 and ALARA and to minimize the potential for health effects. Effective ALARA planning will ensure that adequate physical and administrative controls are in place to mitigate exposure during radiologically significant work. Industry's experience includes frequent events where problems in this area have resulted in unanticipated exposure or a loss of control of the work activity. Specific attention should be given to Planned Special Exposures and exposures to Declared Pregnant Workers owing to the higher risk involved.
OE	Identification and Resolution of Problems/Issues			Incorporated in each applicable inspectable area		

Table 1, Risk Information Matrix No. 1 (continued)

CORNERSTONE	INSPECTABLE AREA	PERFORMANCE INDICATOR	FREQUENCY	HOURS FOR 2-UNIT SITE PER YEAR	LEVEL OF EFFORT	BASES
OE	Radiation Monitoring Instrumentation	None	Annual	30 hrs/ yr	Inspect all those monitors located in areas subject to significant, transient radiological conditions (including inadvertent criticality) during work activities, and that portable instrumentation used to assess radiologically significant areas or work. The hours are based on the current core program as modified by inspection experience. These hours include 6 hours of Identification and Resolution of Problems/Issues to review licensee identified events; 4 hours of walkdown and 20 hours to review monitor calibration, alarm setpoint and maintenance records	Radiological risk (i.e., exposure) to a worker be within the occupational exposure limits defined in 10 CFR Part 20 and ALARA and to minimize the potential for health effects. These monitors identify changing radiological conditions to workers such that actions to prevent an overexposure can be taken. Industry has experienced several events where these monitors were the primary indication that radiological conditions had significantly changed as a result of planned or unplanned activities.
OE	Radiation Worker Performance	None	Annual	20 hrs/ yr	The inspector shall review, at a minimum, all events that occurred over the last 6-12 months, including specific events identified by the resident inspectors, for adverse trends that affect the Occupational or Public Dose cornerstones. The hours are based on the current core program as modified by inspection experience. These hours include 20 hours of Identification and Resolution of Problems/Issues to determine that the licensee effectively identifies and addresses adverse radworker performance trends and that overall performance is consistent with NRC inspection or PI findings. This area does not include walkdowns or events involving access control to HRAs $\geq$ 1000 mrem/hr, VHRAs or airborne areas as they are addressed in other inspection areas	The associated risk is the potential for a significant, unplanned exposure resulting either directly or in part by the failure of a worker to perform a required task owing to poor knowledge or training. Recurrent problems in this area have been identified by the industry as a root or contributing cause in many exposure events and in some events involving the unplanned release of radioactive material to the environment. This is of special concern during outages, when radiologically significant work is often performed by contract staff having varying levels of experience.

Table 1, Risk Information Matrix No. 1 (continued)

Cornerstone	Inspectable Area	Performance Indicator	Frequency	Hours For 2-unit Site per Year	Level of Effort	Bases
PUBLIC EXPOSURE (PE) CORNERSTONE						
PE	Gaseous and Liquid Effluent Treatment Systems	Reportable Release Events	Biennial	30 hr/2 yr	The inspector shall review the calibration and maintenance records for each effluent monitor (including the backup monitors) and each meteorological instrument. Each system modification should also be reviewed excepting those having a minimal impact on system operation. Every event reported via the PI shall also be reviewed. Total inspection hours required were reduced because of the availability of a PI. The hours are based on the current core program as modified by inspection experience. The total hours include 10 hours of Identification and Resolution of Problems/Issues to review the licensee assessments, events and annual effluent and environmental reports; 5 hours to walkdown the gaseous and liquid systems (including monitors) to observe the equipment material condition and ongoing activities; and 15 hours to review effluent and meteorological monitor calibrations, monitor alarm setpoints, maintenance records and system modifications.	Radiological risk (i.e., exposure) to the public be below the 10 CFR Part 20 and 40 CFR Part 190 limits and ALARA to minimize the potential for health effects. Doses below the design objectives of Appendix I to 10 CFR Part 50 are considered ALARA by the NRC. Proper operation of the effluent treatment system and monitors will ensure an adequate "defense-in-depth" against an unmonitored, unanticipated release of radioactivity to the environment. Overall industry performance has improved, but concerns still exist with abnormal releases, system modifications, and monitor operability.
PE	Identification and Resolution of Problems/Issues			Incorporated in each applicable inspectable area		
PE	Radioactive Material Processing and Shipping	None	Annual	40 hrs/ 2 yrs	At a minimum, the inspectors should review 4-5 radiologically significant shipment packages, observe the surveying, placarding, etc for at least 1 shipment, and observe at least 1 radioactive material processing activity (i.e., resin dewatering, waste sorting, waste packaging, etc). The hours are based on the current core program as modified by inspection experience. These hours include 10 hours of Identification and Resolution of Problems/Issues to review licensee assessments and events; 6 hours to observe radiologically significant processing and shipping activities (such as RWCU resin dewatering, Type A or B shipments), and radioactive material work and storage areas; and 24 hours to review associated records including 10 CFR Part 61 sample collection and analysis results. Credit shall be taken for any of these activities observed/reviewed while performing another inspectable area.	The regulations state specific physical and administrative controls that provide for a layered defense against unplanned radiation exposure during radioactive material processing and transport or from an accidental breech of the shipping container. Although there is a low frequency of industry events, the actual or potential consequence (i.e., significant exposures or release of radioactive material) is typically high.

Table 1, Risk Information Matrix No. 1 (continued)

Cornerstone	Inspectable Area	Performance Indicator	Frequency	Hours For 2-unit Site per Year	Level of Effort	Bases
PE	Radiological Environmental Monitoring Program (REMP)	None	Biennial	10 hrs/ 2 yrs	The inspector shall observe, at a minimum, the on-site samplers and TLD monitoring stations, and observe the collection and preparation of either a liquid, vegetation or soil sample. At least 5 samplers (on-site and offsite) shall be selected for review of associated calibration and maintenance results. Each event regarding a missed sample, inoperable sampler or lost TLD documented in the annual report shall also be reviewed. The hours are based on the current core program as modified by inspection experience. These hours include 2 hours for Identification and Resolution of Problems/Issues to review licensee assessments and identified problems; 4 hours to walkdown the sampling stations and observe environmental sample collection and processing; and 4 hours to review counting room instrumentation and environmental sampler calibration and maintenance records. No time is allotted for review of the annual environmental report or meteorological instrumentation as this was included in the Gaseous and Liquid Treatment Systems inspection area.	The REMP supplements the effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are as predicted by the effluent measurements and modeling of effluent pathways. As such, it serves as the final barrier in assuring that the associated dose from radioactive releases is within regulatory limits.

Table 1, Risk Information Matrix No. 1 (continued)

Cornerstone			Inspectable Area	Performance Indicator	Frequency	Hours For 2-unit Site per Year	Level of Effort	Bases
REACTOR SAFETY CORNERSTONES			(I=Initiating Events; M=Mitigating Systems; B=Barrier; X or a number indicates the Inspectable Area is mapped to that Cornerstone; When a number is present in a column, it represents the approximate percentage of the total hours of inspection to be performed for that Cornerstone)					
I 20	M 80	B	Adverse Weather Preparations	None	As conditions require	12 to 18 hrs/year	Select 1 non-failure tolerant SSCs, supplemented by 1 site-specific high risk SSCs. The non-failure tolerant SSCs (i.e., highly reliable RWST), whose failures may contribute a small amount to the total CDF, but create a large CCDP, could result in failures of other SSCs due to instrument line freezing or other CCF failures. Use plant history, IPE, IPEEE to determine vulnerability and assign final hours. Baseline inspection to be performed prior to seasonal susceptibilities. Hours include 6 hrs for Identification and Resolution of Problems/Issues.	Conditions leading to Loss of Offsite Power, freezing temperatures, high winds, flooding dominate risk. Conditions can lead to common cause failure of mitigation equipment and to initiatiating events.
I	M 80	B 20	Changes to License Conditions and Safety Analysis Report	None	Annual	32 hr/yr	Review licensee evaluations made per 10CFR50.59 requirements. If the initial screening indicates that the issues potentially increase risk, select the issue for review. Select a minimum of 5 significant evaluations for indepth review. Includes 8 hrs of Identification and Resolution of Problems/Issues	Changes can be made without prior NRC approval only if they do not increase risk. Adequate licencee performance while evaluating impact of changes prevents changes that increase risk from being made. Success criteria for PRA could change if license basis changes.
I 40	M 60	B	Emergent Work	None	Bimonthly	60 hrs/yr	Selection of risk significant activities should be made using licensee's configuration specific risk assessment or from a ranking of system importance. RIM2 should be used if plant specific information has not yet been developed. Select 2 activities per month. Hours estimate assumes 3 hrs/month of observation and 2 hr/month of Identification and Resolution of Problems/Issues.	Troubleshooting while trying to determine cause of emergent equipment problems can lead to inadvertent risk significant initiating events. In addition, high risk configurations with multiple out-of-service SSCs may occur during rolling on-line maintenance due to emergent work.
I 30	M 60	B 10	Equipment Alignment	None	Semiannual and as required by maintenance	76 hrs/yr	One system walkdown every 6 months. If available system success criteria from the site specific risk study, and the system design basis should be reviewed to focus the inspection. RIM2 should be used for system selection if plant specific information has not yet been developed. In conjunction with maintenance on higher risk systems, validate critical features on lineup of the train or system providing the backup function. Hours based on 8 hrs semiannually for a complete risk important system walkdown; 4 hrs/month in walkdowns to support verification of operable system train because other train is OOS, and 1 hr/month for Identification and Resolution of Problems/Issues.	High risk configurations may occur during normal operations and on-line maintenance activities due to multiple out-of-service SSCs, and such configurations can lead to high Core Damage Probability.

Table 1, Risk Information Matrix No. 1 (continued)

Cornerstone			Inspectable Area	Performance Indicator	Frequency	Hours For 2-unit Site per Year	Level of Effort	Bases
I 10	M 90	B	Fire Protection	None	Triennial	36 hours/3 yrs 12 hr/yr Residents	Selection of areas inspected should consider insights from the plant specific fire risk analysis. Regional SRA to provide input. Walkdown all accessible areas of high significance. Hours are based on a regional based Program Implementation Review, and 4 hours of Identification and Resolution of Problems/Issues. Residents should perform a monthly walkdown of high fire risk areas (hours based on One hr/walkdown) to verify transient combustible loading and fire doors/barriers.	Estimated fire risk is comparable to many internal initiating events. If potential fire initiators, aids to propagation, or fire barrier breaches exist, safe shutdown of the plant may not be possible due to the failures of the inspectable features and areas.
I 40	M 60	B	Flood Protection Measures	None	Annual	20 hrs/yr	Internal and External flood protection barriers and actions review. Select from both the external and internal flooding scenarios, perform walkdown of areas, being sensitive to insights from the plant specific flood risk studies. Includes 4 hrs/yr of Identification and Resolution of Problems/ Issues.	IPE and IPEEE summaries indicate that CDF contributions by internal flooding is generally one order of magnitude higher than that by external flooding (less than 1.0E-5 CDF) for most BWR and PWR plants except BWR 1/2/3. At some sites flooding can be a significant contributor to risk.
I	M	B X	Fuel Barrier Performance	Reactor Coolant Activity	Quarterly review of PI	1 hr/yr	Total inspection hours required were reduced because of the availability of a PI. Verification of PI may be required periodically. With review of PI, also check for reportable event involving exceeding core thermal limits or margins.	Fuel cladding is the first barrier.
I 20	M 80	B	Heat Sink Performance	None	Annual Biennial	6 hrs/yr Resident 24 hrs/2 years Regional Specialist	Once a year, for heat exchangers and heat sinks in risk important systems, observe periodic performance testing with a focus on the potential impact of common cause failures. Exchanger selection should be made focusing on high risk functions on exchangers that have low margin to their design point, or have potential for high fouling. One activity per yr for 5 hours. One hr of Identification and Resolution of Problems/Issues. Biennial inspection on heat sink performance conducted by regional specialist. Includes 12 hrs/2 yrs of Identification and Resolution of Problems/ Issues.	Heat exchangers and heat sinks are required to remove decay heat, and provide cooling water support for operating equipment. Degradation in performance can result in failure to meet system success criteria, and lead to increased risk primarily due to common cause failures.
I 10	M 70	B 20	Identification and Resolution of Problems/Issues	All Reactor Safety Performance Indicators	Biennial Daily	120 hrs/yr (240/2 yr) by Regional Specialists Additional hours incorporated in each applicable inspectable area	Region based inspection every two years using 240 hrs. SRA should brief team on site specific risk study and provide insights to aid in selection of items to inspect in the review. Select one or two systems depending on scope of inspection, and complexity of selection. For the review by inspectors, selection should not necessarily be based on individual system importance, but should look at the potential impact of the root cause on the plant as a whole.	Uncorrected root causes to problems could lead to increasing common cause and human event rates, and to breakdowns in multiple Cornerstone areas. Many of the issues will cross multiple Cornerstones.

Table 1, Risk Information Matrix No. 1 (continued)

Cornerstone			Inspectable Area	Performance Indicator	Frequency	Hours For 2-unit Site per Year	Level of Effort	Bases
I 50	M	B 50	Inservice Inspection Activities	RCS Leak Rate	Annual	24 hrs/yr	Total inspection hours required were reduced because of the availability of a PI. Regional based specialist to perform. Activity may be performed on a refueling cycle basis to observe activities. Review or inspection of steam generator tube non-destructive examinations would be performed following each refueling outage. For Welding ISI select a minimum of 3 welds for review. Includes 4 hrs/yr of Identification and Resolution of Problems/ Issues. Review or inspection of reactor vessel non-destructive examinations would be performed at the appropriate 10 year interval. Inspection activities related to non-code repairs would be specified when non-code repairs are performed. SRA will provide input to select repairs to inspect.	ASME Class 1, 2, & 3 components have relatively high reliability components. However, they are non-failure tolerant SSCs and their failures could result in a high Conditional Core Damage Probability and consequences.
I	M X	B	Inservice Testing	None	Bimonthly	64 hrs/yr	Select 2 tests per month. RIM2 should be used for component selection if plant specific information has not yet been developed. Selection will also be based on the history of previous licensee implementation problems in this area, any adverse trends identified within the Section XI pump or valve trending program, and/or following significant maintenance or modification activities on specific components. For valve testing, select samples from different valve groups to increase the inspection's sensitivity to common cause failures. Hours assumes observation of 24 tests on high risk components with associated Identification and Resolution of Problems/Issues time.	Inservice testing provides indication of equipment availability and reliability. Improper testing could result in undisclosed problems that last until the next required testing, unless discovered through failure while in service earlier, creating long periods of unknown equipment inoperability.
I	M	B X	Large Containment Isolation Valve Leak Rate and Status Verification	Containment Penetration Leakage	Status-monthly Leakage-refueling	8 hrs/yr	At PWRs, monthly verify hours purge valves were open, look for increasing trends. At refueling intervals observe one LLRT for high impact valve with large flexible seat area. Total containment leakage inspection hours required were reduced because of the availability of a PI. Includes 2 hrs/yr of Identification and Resolution of Problems/Issues and configuration control on large containment valves that are frequently cycled.	Large containment valves that are cycled open during plant operation increase the likelihood of failure of the containment barrier. Frequent operation and aging of seals in the valves can lead to excessive leakage rates.
I	M 75	B 25	Licensed Operator Requalification	None	Annual Quarterly	96 hrs/year by Regional Specialist 8 hrs/yr by Resident staff	Assumes 96 hours onsite by a regional based specialist. Regional activities based on the required program. Insure licensee includes training on high risk operator actions based on SRA input or RIM2. Does not include in office review of tests performed by the regional specialists. Resident review should focus on the high risk operator actions from the site specific risk study. Assumes 2 hours of simulator activities sampled once per quarter by the Resident inspectors.	Human errors and failure to recover from accident events increase the significance of important events: Examples include failure to manually depressurize and failure to recover offsite power for BWRs, and failure to switch from RWST to containment sump for PWRs.

Table 1, Risk Information Matrix No. 1 (continued)

Cornerstone			Inspectable Area	Performance Indicator	Frequency	Hours For 2-unit Site per Year	Level of Effort	Bases
I 10	M 80	B 10	Maintenance Rule Implementation	None	Annual Monthly	40 hrs/yr for Regional Specialist 216 hrs/yr for Resident Staff	Selection for annual review should be made with input from SRA. Includes a 40 hr annual review by a region based specialist; 16 hrs/month of resident inspector activities, and 2 hrs /month of Identification and Resolution of Problems/Issues. Inspectors should focus on categorization of failures used in tracking condition of important systems, and goal setting and get well program for risk significant A1 Systems. Residents sample 2 systems per month.	Tracking and documenting system availability and reliability for the plant's risk important systems is performed under the Maintenance Rule. These estimates impact the plant risk model, in addition to actual plant risk.
I 10	M 80	B 10	Maintenance Work Prioritization and Control	None	Monthly	34 hrs/yr	Select one sample per month for times when multiple component outages were planned simultaneously, or for those times when planning decisions were made for expediting equipment return to service because of component failures, especially on the backshift when normal planning was unavailable. Use site specific risk tools, if available. Prior to planned outages, review the outage plan, and its risk evaluation. Includes 4 hrs/year of Identification and Resolution of Problems/ Issues.	Control of plant risk and configurations through appropriate planning and control of maintenance activities minimizes the plant's aggregate risk. Work prioritization and risk evaluation prior to outages minimizes risk significant configurations and maximizes barriers to radiological release.
I X	M X	B X	Nonroutine Evolutions	None	As Required	102 hrs/yr (Hour allocation between Corner-stones will be based on actual events; assume even distribution for planning purposes)	All major evolutions should be considered for review and/or observation. The items to be selected should be based on the complexity of the activity and the potential risk significance of possible operator errors and/or equipment problems. Includes 40 hrs/year of Identification and Resolution of Problems/ Issues. Hours based on six occurrences per year of risk significant off normal operation, one post-scram review, and 10 risk significant LER reviews per year.	Human errors, particularly recovery actions from event initiations, are the major contributors to plant risk. Performance during non-routine operations can be used as an indicator of plant personnel performance during emergencies than their performance during normal operations. In addition, plant upset events are more likely during nonroutine operations.
I	M X	B	Operability Evaluations	None	Monthly	60 hrs/yr	Review all operability evaluations to identify those involving systems or components with the greatest impact on plant risk. Perform a more in depth review of all risk significant evaluations. RIM2 should be used for selection if plant specific information has not yet been developed. Includes 12 hrs/year of Identification and Resolution of Problems/ Issues. Hours based on 2 activities per month of 2 hrs each.	Inoperability of components can result in high risk configurations because the tools being used to evaluate risk in daily planning will have invalid equipment availability.
I	M X	B	Operator Work-Arounds	None	Monthly	30 hrs/yr	Review operator workarounds monthly, and select two to evaluate. Select those items which can impact operator response during events. Based on 2 activities per month of 1 hr each. Includes 6 hr/year of Identification and Resolution of Problems/ Issues.	Operator workarounds can impact human performance during event response, due to increasing complexity of tasks and more limiting time to perform required actions. .

Table 1, Risk Information Matrix No. 1 (continued)

Cornerstone			Inspectable Area	Performance Indicator	Frequency	Hours For 2-unit Site per Year	Level of Effort	Bases
I	M 80	B 20	Permanent Plant Modifications	None	Annually At time of occurrence, if high risk and performed on line	80 hrs/yr regional specialist 24 hrs/yr Resident Staff	Review plant-specific configuration risk for pre- and post-modification, and potential unreviewed safety questions. Select 3 to 5, depending on complexity, of those modifications with the greatest impact on risk. Regional inspectors should use the SRA to input into the selection of modifications to be reviewed. Hours are based on 5 modifications at 16 hours of review per modification by a Regional specialist. Includes review of post modification testing. Includes 8 hrs/year of Identification and Resolution of Problems/ Issues. Resident inspectors will select an activity based on its importance to risk, and complexity. Use the RIM2, if plant specific information has not yet been developed. On site inspection by either the resident or regional based inspector for configuration and post mod test review of ongoing modifications will require an additional 20 hours.	Plant modifications will impact plant risk, by either increasing or decreasing the baseline risk after the modification. The risk changes are expected to be small, however the concern is an increase in risk and/or consequence during the modification, if performed on line.
I X	M	B	Piping System Erosion/Corrosion	None	Refueling	16 hrs/yr	16 hrs/refueling review/ observation; 8 hrs/refueling Identification and Resolution of Problems/Issues. Performed by regional specialist. Select sample from both the erosion and the corrosion program.	High energy and high risk system piping breaks are relatively low risk but an actual failure may result in a significant event, or high Conditional Core Damage Probabilities.
I	M X	B	Post Maintenance Testing	None	Monthly	72 hrs/yr	Selection should focus on high risk components. RIM2 should be used if plant specific information has not yet been developed. Select an average of two per month. Hours based on review of 2 risk significant activities/month, and 3 hrs/activity. Inspector should determine if scope of testing adequately covered the work performed, and final equipment configuration.	Inadequate PMT could result in unrealized inoperability of equipment for extended periods of time, and could impact any risk based planning tools ability to model plant risk. This could result in high risk situations when other equipment is taken out of service.
I 20	M 70	B 10	Refueling and Outage Related Activities	None	Refueling	80 hrs/yr	Includes review of mid-loop ops for PWRs. Licensee's outage risk assessment to guide inspectors to risk significant activities. Performed on outage basis, not annual. Inspection should focus on: RHR, Containment isolation during reduced water inventory, Mid-loop (PWRs) operation, cool down/heatup/startup, availability of alternate power sources/switchyard, and Refueling operations. Includes 30 hrs/refueling of Identification and Resolution of Problems/ Issues.	Shutdown risk will be high if vital SSCs are not available. Due to potentially high numbers of out-of-service SSCs during the refueling period, configuration risk can be high. Times of reduced inventory are the most critical.
I	M X	B	Safety System Design and Performance Capability	None	Biennial	120 hrs/yr	System selection with input from regional SRA. Select one or two systems, depending on complexity. Region based biennial risk important system design review. 240 hrs/activity.	Functionality of high risk significant SSCs are verified, including design basis, support functions, installation, testing, normal/emergency functions. Factors contributing to risk reduction/increase are verified through validation of the success criteria.

Table 1, Risk Information Matrix No. 1 (continued)

Cornerstone			Inspectable Area	Performance Indicator	Frequency	Hours For 2-unit Site per Year	Level of Effort	Bases
I	M 90	B 10	Surveillance Testing	Safety System Performance Indicators	Monthly	48 hrs/yr	Select systems surveillance that are performed on systems ranked high in importance in the site specific risk study, or if not available, from RIM2. Inspect an average of 2/month. Total inspection hours required were reduced because of the availability of a PI. Assumes 2 risk-significant activities at 4 hrs per month. Includes 12 hrs/year of Identification and Resolution of Problems/ Issues.	Provides indication of system operability. Plant configuration and system restoration are important.
I	M 90	B 10	Temporary Plant Modifications	None	As Required	28 hrs/ yr	Screen for temporary modifications with relatively high risk configurations. RIM2 should be used if plant specific information has not yet been developed. Includes 4 hrs/year of Identification and Resolution of Problems/ Issues.	The modifications may result in a departure from the design basis and system success criteria, and can result in a configuration that may be an unreviewed safety concern. Temporary or unrecognized risk changes due to the modification may evolve into high risk configurations

Table 1, Risk Information Matrix No. 1 (continued)

CORNERSTONE	INSPECTABLE AREA	PERFORMANCE INDICATOR	FREQUENCY	HOURS FOR 2-UNIT SITE PER YEAR	LEVEL OF EFFORT	BASES
SECURITY (SEC) CORNERSTONE						
SEC	Access Control	None	Annual	24 hrs/yr	Perform the following: Observe the licensee conduct performance based testing of all access control equipment using their testing methods and devices. Observe the processing of personnel, packages, and vehicles entering the protected area to determine if security can effectively perform all of the tasks assigned to them by procedures. Verify that keys to vital areas are controlled and protected in accordance with requirements. Expert judgement basis for hours. 16 hrs per year to perform Access Control Testing 8 hrs per year to perform Identification and Resolution of Problems/Issues	Failure of program compromises security barriers in place to protect high risk plant equipment and activities. Risk consequence to radiological sabotage is moderate.
SEC	Access Authorization Program	Number of Reportable Access Authorization Events	Annual	12 hrs/yr	Performance will be verified by reviewing audits and quarterly/semiannual data submitted to the NRC. The Behavior Observation significant events are captured in the FFD reporting requirements. However, a minimum baseline inspection should be conducted of the Behavioral Observation program process and human performance attributes . Expert judgement basis. PI reduces the inspection effort, and covers all areas except Behavioral Observation Program process and PI verification. Two hours for Identification and Resolution of Problems/Issues are included.	Failure of program compromises ability to protect against the insider threat of radiological sabotage. The consequences due to radiological sabotage can be high if program fails.
SEC	Changes to License Conditions and Safety Analysis Report	None	As required	8 hrs/yr	In office review all Security Plan change submittals made per 10 CFR 50.54 requirements. Hours based on expert judgement for 2 plan changes per year.	Changes can be made without prior NRC approval only if they do not decrease plan effectiveness. Adequate licensee performance while evaluating impact of changes prevents changes that could increase the likelihood of damage due to radiological sabotage.
SEC	Identification and Resolution of Problems/Issues					
SEC	Physical Protection System (Barriers, Intrusion Detection System, and Alarm Assessment)	Availability and capability of security equipment	Quarterly review of PI	8 hrs/yr	In office review of PI results only. Total inspection hours required were reduced because of the availability of a PI. Regional desktop review of reports. 2 hrs/quarter per facility.	Risk significance is based on an exploitable vulnerability by a person(s) with the intent and capability to commit radiological sabotage. The risk consequences of such an event would be moderate to high.

Table 1, Risk Information Matrix No. 1 (continued)

CORNERSTONE	INSPECTABLE AREA	PERFORMANCE INDICATOR	FREQUENCY	HOURS FOR 2-UNIT SITE PER YEAR	LEVEL OF EFFORT	BASES
SEC	Response to Contingency Events (Protective Strategy and Implementation of Protective Strategy)	None	Biennial	104/2 yrs	Conduct and Evaluate Security Exercise. Three inspectors. Includes onsite inspection of Physical Protection as part of exercise. Expert judgement basis for hours.	This is a high risk-significant function necessary to protect against the design basis threat of radiological sabotage. The risk consequence to radiological sabotage if a successful attack did occur is high.

References for Table 1, RIM No.1

1. Chung, J.W., Travis, R., etc., “Generic Risk Insights for Westinghouse and Combustion Engineering Pressurized Water Reactor”, Brookhaven National Laboratory, NUREG/CR-5637

2. Chung, J.W., Travis, R., etc., “Generic Risk Insights for General Electric Boiling Water Reactors”, Brookhaven National Laboratory, NUREG/CR-5692

3. Chung, J.W., Vesely, W.E., Thadani, A.C., “Risk Management Strategies - Qualitative and Quantitative Approaches”, presented in PSA‘95 International Conference, Probabilistic Risk Assessment Methodology and Applications.

4. Chung, J.W., Wong, S., Riley, J.E., “Risk Profile Methodology of Plant Configuration and Pilot Applications: Lessons Learned”, Draft NUREG-1605.

5. “Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance”, NUREG-1560

6. “Preliminary Perspectives Gained from Initial Individual Plant Examination of External Events (IPEEE) Submittal Review, Draft NUREG.

7. Vesely, W.E., “A Systematic Process For Risk-Prioritizing Inspection Activities”, presented to USNRC on December 8, 1998, Contract Report.

8. Vesely, W.E., and Davis, T.C., “Evaluations and Utilizations of Risk Importances”, Battelle Columbus Laboratories. NUREG/CR-4377.

9. Vesely, W.E., and etc., “Measures of Risk Importance and their Applications”, NUREG/CR-3385

Table 2, Risk Information Matrix No. 2: PWRs

PWR Systems that Were *Risk Important at Most Plants* and Reasons for Importance Not prioritized within this portion of the Table				
Important Systems from IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Systems selected based on CDF contribution from NUREG-1560				
Offsite Power	The loss of offsite power is an important initiator event, if lasting beyond 30 minutes. Loss of offsite power could occur as a result of adverse weather conditions, switch yard activities, degraded grid, component failure, design inadequacy, or human error (e.g., when maintenance on part of system is being performed). Losses of offsite power due to adverse weather conditions could be important since the harsh weather condition such as freezing could impact other mitigating systems. Some losses of offsite power could be recovered by operator actions. Recovery of offsite power is shown to be important in PRAs.	X	X	
Emergency AC	Common cause failures of multiple emergency diesel generators has been a major risk concern. The unavailability of one diesel generator either due to failure or maintenance, combined with the failure of the other safety trains that are fed from the remaining emergency bus, have also been a major contributor to risk. Therefore, configuration control would be important during maintenance or emergent work on EDGs. Recovery of Emergency AC or EDGs in SBO sequences is important.		X	X
RCP Seals	In the event of an SBO, the failure of RCP seals due to loss of seal cooling (CCW) and injection (charging system) would induce a LOCA. Maintaining or quickly recovering cooling to the seals is important. RCP seal failures in Westinghouse pumps during normal operation could result in a small LOCA initiator. Such events have occurred in the past. Improper installation of seal cartridge, shaft eccentricity, and injection flow instabilities are some of the failure causes.	X		X
DC Batteries	The plant I&C and the turbine driven AFW are fed from DC buses. During SBO scenarios, DC load shedding is important to extend battery depletion time. This provides more time for recovery of AC power in an SBO sequence. During normal operation total or partial loss of DC system could be a major risk contributor and in some plants could cause severe transients and loss of decay heat removal.	X	X	X

Table 2, Risk Information Matrix No. 2: PWRs (continued)

PWR Systems that Were *Risk Important at Most Plants* and Reasons for Importance Not prioritized within this portion of the Table				
Important Systems from IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Systems selected based on CDF contribution from NUREG-1560				
Low Pressure Injection (LPI)	LPI/LPR system is important in LOCA sequences. A major concern in PWRs is the failure of operators to properly switch to the recirculation mode. In some plants this would be done automatically, therefore is of lesser concern. In an ice condenser containment, the switch over could be more important due to shorter time period involved. Another important issue related to the LPI system deals with a LOCA due to failure of high and low pressure interface. A series of interface check valves, isolate the RCS from the LPI system. Large back-leakages through a pair of these interface check valves result in an interfacing ('V" sequence) LOCA. Three major concerns are: the capability to isolate through closure of MOVs (design) ,the survivability of the remaining intact train (EQ), and the operator actions.	X	X	X
PORVs	Feed and Bleed is the option of last resort in removing decay heat. Reliability of PORVs and block valves and the associated human actions for feed and bleed are important in this mode of operation. Depending on the plant design, either one or both PORVs would be required for feed and bleed. Both PORVs and Block valves require either DC or AC power. Therefore, in loss of vital AC or loss of DC scenarios, the PORVs may not be credited. In some plants, feed and bleed may not be possible due to the low head of safety injection pumps. Feed and bleed capability usually does not exist in CE plants. In transients, the failure of PORV to re-close results in a LOCA scenario Spurious actuation of PORV due to fire is also reported in some fire PRAs.	X	X	X

Table 2, Risk Information Matrix No. 2: PWRs (continued)

PWR Systems that Were *Risk Important at Most Plants* and Reasons for Importance Not prioritized within this portion of the Table				
Important Systems from IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Systems selected based on CDF contribution from NUREG-1560				
SGs	SG atmospheric dump valves (ADVs) are important as a means of rapid heat removal and an alternate way to depressurize the RCS during a small LOCA with loss of HPI. During transients, the ability to depressurize the SGs and the use of the condensate system for heat removal is quite important for CE plants, where the feed and bleed capability does not exist. Flooding of steam generators after core damage is an important accident management strategy which results in scrubbing of radionuclides and reducing the public health effects. SGTR is a failure of the RCS pressure boundary and is an important containment bypass mode for most PWRs.	X	X	X
RWST	Important for LOCA mitigation. The size of the RWST and the appropriate maintenance of the water level plays an important role in determining the time available for switch over to the re-circulation Phase. Means for replenishing the water in RWST are also important.		X	
HPI/HPR	The HPI/HPR system could be used as a mitigating system for LOCAs and as a means of cooling through feed and bleed during transients. During the injection phase, the common cause failures of injection discharge valves and multiple failures or combined failures and maintenance unavailabilities of the pumps are the major contributors. During the recirculation phase; the loss of pump cooling, human error associated with switch over, and containment sump clogging are the major contributors.		X	

Table 2, Risk Information Matrix No. 2: PWRs (continued)

<table>
<tr><th colspan="5">PWR Systems that Were Risk Important at Most Plants and Reasons for Importance
<u>Not prioritized within this portion of the Table</u></th></tr>
<tr><th rowspan="2">Important Systems from IPEs</th><th rowspan="2">Reasons for Importance</th><th colspan="3">Cornerstones</th></tr>
<tr><th>Initiating Events</th><th>Mitigating Systems</th><th>Barrier</th></tr>
<tr><th colspan="5">Systems selected based on CDF contribution from NUREG-1560</th></tr>
<tr><td>AFW/EFW</td><td>Important for decay heat removal on transients and sequences initiated by full or partial loss of AC or DC power (e.g. SBO sequences). Initiation of AFW and providing an extended source of water (for example from fire water) are considered important in PRAs.

AFW/EFW could fail due to system interaction and common cause failures. Common cause failure (CCF) induced by steam binding as a result of leakage from main feed water through the AFW pump discharge check valves, which flashes to steam in AFW pump, is one example. Undetected flow diversion, for example through the cross connections, is another system failure scenario.

The turbine driven AFW pump could have large unavailability due to maintenance. Other contributors to failure of AFW pumps are local failures of the steam admission line, and of the turbine driven pump.</td><td></td><td>X</td><td></td></tr>
<tr><td>Cross-ties</td><td>Cross-ties between systems and units will provide more redundancy and therefore are important to plant safety. However, in some plants these cross connections have been shown to result in increased chance for diversion of flow, and increase in complexity of human actions.</td><td></td><td>X</td><td>X</td></tr>
<tr><td>Instrument Air System</td><td>The loss of instrument air could not only cause an initiating event such as MSIV closure, but also could impact the operation of many mitigating systems including AFWS, PORVs, SG relief, and Boron injection. It is also important to note that in some plants loss of instrument air could impact the HVAC system which is shown to be risk significant.</td><td>X</td><td>X</td><td></td></tr>
<tr><td>Service Water</td><td>Dependency of plant systems on SW (both normal and emergency) is important. The SW system is typically the heat sink for the CCW system, however there is some cooling capacity in CCW even if SW is lost. In addition to the systems cooled by CCW, loss of SW could affect AFW pumps (both motor- and turbine-driven), EDGs, and HVAC systems.</td><td>X</td><td>X</td><td>X</td></tr>
<tr><td>CCW</td><td>Loss of CCW is quite important in several PWRs, since it impacts several mitigating systems and could cause LOCA, as a result of RCP seal leakage. Dependency of plant systems on CCW is important. The CCW system is also dependent on support systems (AC power and SW).</td><td>X</td><td>X</td><td>X</td></tr>
</table>

Table 2, Risk Information Matrix No. 2: PWRs (continued)

PWR Systems that Were *Risk Important at Most Plants* and Reasons for Importance **Not prioritized within this portion of the Table**				
Important Systems from IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Systems selected based on CDF contribution from NUREG-1560				
Containment Isolation	Containment isolation failures are important at many PWRs (e.g., large dry and sub-atmospheric containments). Early failure of containment prior to core damage or shortly after core damage could result in a LERF. Failure of high/low pressure interface valves is an important containment bypass mode (see the discussion under LPI). SGTR is another mechanism for containment bypass, however of lesser consequence, if the SG is filled with water (see the discussion under SG).	X		X
Containment Sump	The water in the sump after a LOCA is the main source for the recirculation phase of the ECCS. Sump clogging either due to pieces of broken thermal insulation around the primary pipes or from foreign materials left inside the containment during refueling (e.g., plastics) could be an important risk contributor.		X	
HVAC System	Loss of HVAC system in some areas such as switchgear rooms, could cause a loss of AC power. A major cause for loss of DC in some plants was loss of the DC equipment HVAC system. Among other ways, loss of HVAC could occur as a result of closure of fire dampers after a fire test. Availability of early detection systems, such as temperature alarms, would be important for the timely restoration of the HVAC. HVAC systems require support systems to operate. One support system of concern is the instrument air system.	X	X	X

Table 2, Risk Information Matrix No. 2: PWRs (continued)

Additional PWR Items that Were *Risk Important at Most Plants* and Reasons for Importance				
Important Systems from IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Additional Systems – Important based on high RAW (note 2)				
RPS	Reactor protection system for preventing ATWS (in some plants identified by RAW and some by FV). Failure of the reactor protection system results in ATWS scenarios that are considered important in some plants. Failure of reactor protection system is dominated by the CCF of the scram breakers and mechanical failure of the control rods. For plants with a diverse rod insertion system, high unavailability of the system is a secondary contributor.		X	
Primary System Integrity	The failure of primary pipes including reactor vessel integrity is a low probability event with a high consequence.	X		X
Shutdown Risk Insights				
Normal AC Power	Loss of offsite power is an important initiator in shutdown risk. Duration of loss of offsite power which is risk significant varies depending on the plant operational state and the decay heat. As an example, It could vary from 40 minutes for hot shutdown to as high as 1.5 hr for mid loop operation. Loss of offsite power frequency could be higher during shutdown because of surveillance and maintenance activities of electrical equipment, switch yard, and potential degradation of the grid.	X		
Emergency AC	SBO is a dominant risk contributor during shutdown specially in mid-loop operation. Loss of emergency AC, with loss of offsite power, is considered as a major initiator to shutdown risk.	X	X	
RHR	Loss of RHR during mid-loop operation is a primary contributor to shutdown risk. It is considered as one of the primary initiators during shutdown.	X	X	
RCS Pressure Boundary	Introduction of weaknesses into RCS pressure boundary has been found to be a potential problem. Possible weaknesses include thimble tube seals, steam generator nozzle dams, and freeze seals.	X		X

Table 2, Risk Information Matrix No. 2: PWRs (continued)

Additional PWR Items that Were *Risk Important at Most Plants* and Reasons for Importance				
Important Systems from IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Configuration Management	An important issue in shutdown risk relates to shutdown configuration management and control. Utilities have used different methods for assuring the configuration control with varying degree of effectiveness. Some utilities have a minimum requirements list, which specifies equipment by function that should be available. Some other utilities have a comprehensive ORAM (Outage Risk Assessment and Management Plan). The adequacy of the methods used and the operator's understanding of plant configuration is shown to be important in shutdown.		X	X

Table 2, Risk Information Matrix No. 2: PWRs (continued)

Additional PWR Items that Were *Risk Important at Most Plants* and Reasons for Importance				
Important Systems from IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Shutdown Risk Insights (continued)				
Reactivity Accidents	Reactivity accidents, mainly as a result of Boron dilution, are suspected to be major contributors to risk. However, currently there is some uncertainty in this regard.	X		
RCS Pressure Relief	The pressure relief capability when RHR is lost is important. Primary pressurization occurs as a result of thermal expansion of the coolant or steaming in the reactor vessel on loss of RHR.		X	X
Level Instruments for Mid-loop	Inadvertent draining during mid-loop operation could occur either by human errors or by equipment failures. Reliable level instrumentations would help to prevent or detect inadvertent draining initiators.	X	X	
Fire Protection Insights				
Switchgear Room	A fire in the emergency switchgear room in some plants could fail cabling for CCW and HPI systems therefore causing an RCP seal LOCA. In other plants, this could result in the loss of the ESW system, which fails the diesel generator (the only source for the AC power), therefore causing an SBO. The SBO scenario could become more severe upon loss of RCP seal cooling, inducing RCP seal LOCA.	X	X	X
Cable Spreading Room	A fire in the cable spreading room in some plants could be a significant contribution to risk. It would follow a scenario similar to that of control room fire.	X	X	X
Cable Vault and Tunnel	A fire in cable vault and tunnel could fail cabling for CCW and HPI systems therefore causing RCP seal LOCA and also damaging mitigating systems.	X	X	X
Control Room	A fire in control room can force evacuation of control room and in some plants could cause spurious actuation of PORVs with subsequent failure to re-close. It should be noted that once the control room is abandoned, there is typically no independent indication of the PORV position.	X	X	X
Fire Protection Insights (continued)				

Table 2, Risk Information Matrix No. 2: PWRs (continued)

Additional PWR Items that Were *Risk Important at Most Plants* and Reasons for Importance				
Important Systems from IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Fire Protection Systems	The three general areas to address fire risk are: prevention, mitigation/suppression, and safe shutdown capability. NUREG/CR-4230 found that the following fire protection system features were the most important to risk: qualified cables, rated barriers, automatic suppression, and automatic doors and dampers.	X	X	
Internal Flooding Insights				
Buildings with Potential for Flooding	The type of equipment, layout, elevation, adjacency to the water sources and their piping are major consideration in identifying the buildings that are susceptible to flooding and could contribute to plant risk. This issue requires plant specific treatment.	X	X	X
Sources of Flood	The IPEs have identified the following sources of internal flooding: Circulating water system, Fire water system, Service water system, Component cooling water system, Auxiliary feed water system, Main feed water system, and intake structure.	X	X	X
Human Actions				
Restoration of Room Cooling	In scenarios involving loss of HVAC system, the room cooling can be re-established either by recovery of HVAC or opening doors and utilizing portable fans.	X	X	X
Establishing Recirculation	In LOCA scenarios the switching of ECCS lines from injection to recirculation is done manually. Failure to do so or human error involving valve alignment is important.		X	
Feed and Bleed	Failure of the operator to initiate and perform the feed and bleed operation as a last resort of heat removal.		X	
Water Supply for AFW	Use of water pumps to transfer water, from other sources of make up to the CST, is considered important in scenarios when long term cooling through SG is needed.		X	
Human Actions (continued)				

Table 2, Risk Information Matrix No. 2: PWRs (continued)

Additional PWR Items that Were *Risk Important at Most Plants* and Reasons for Importance				
Important Systems from IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Extending the Battery Duration	In SBO scenarios, the operator could extend the duration of the availability of DC by load management to assure the availability of turbine driven AFW pump and the necessary instrumentation and control. This human action is considered important in most PRAs.		X	
Recovery of Emergency AC or Offsite Power	Some losses of AC power could be recovered by either manual transfer of the source of power, or recovery of onsite normal/emergency AC power. This recovery action is considered risk significant in many PRAs.		X	X
Shutdown Operation	Almost all actions, including actuation of various equipment, would be done manually during shutdown. The operator's understanding of the plant configuration is necessary for the successful manual actions.	X	X	X

Table 2, Risk Information Matrix No. 2: PWRs (continued)

PWR Items that Were *Potentially Important* and Reasons for Importance **Not prioritized within this portion of the Table**				
Important Systems From IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Systems selected based on inclusion in several plants in the IPE Data Base				
Boron Injection System	Boron injection is a necessary action during ATWS scenarios. In some CE plants Boron injection was also required during some SGTRs when the fast cooling of the secondary through ADVs occurs. In some plants Boron injection valves are integral part of charging and high pressure injection lineup. Depending on plant specific design, the major concerns are: failure to manually actuate the boron injection, failure of a single MOV to open, and (of a lesser concern) the boron precipitation.		X	
DC System	Total or partial loss of the DC system could cause severe transients and affect the operation of the mitigating systems, including loss of decay heat removal. The impact of partial loss of DC varies significantly among plants.	X	X	X
Vital AC System	Total or partial loss of the vital AC system could cause severe transients and affect the operation of the mitigating systems. The impact of partial loss of vital AC varies from plant to plant.	X	X	X
Containment Heat Removal (CHR)	In some PWRS loss of CHR leads to early over-pressurization of containment and its failure in LOCA scenarios. If this occurs prior to core damage, it would result in flashing of water in the containment sump and loss of ECCS in recirculation phase, which could result in core damage at a later time. Loss of CHR could also result in late containment failure after core damage has occurred.		X	X
MSIVs	MSIVs are relied on to isolate during a steam line break, and SGTR. Spurious closure of MSIVs causes transients with loss of PCS. Failure of MSIV to perform the above functions has been a major contributor in some plants.	X	X	
Primary Safety Relief Valve	PSRVs are relied on in ATWS scenarios to open and limit the peak primary pressure early in the scenarios. Set point drift is not of concern but rather the failure of the valve to open is considered important.		X	X

Table 2, Risk Information Matrix No. 2: PWRs (continued)

PWR Items that Were *Potentially Important* and Reasons for Importance **Not prioritized within this portion of the Table**				
Important Systems From IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Systems selected based on inclusion in several plants in the IPE Data Base (continued)				
Condensate Storage Tank	As a source of water to the AFWS, CST failure could be significant in some plants. In some plants the means to replenish water to the CST, using fire water, is considered as an important risk reduction strategy.		X	
Human Actions				
Make up to RWST	In some W3-loop plants, credit is given for make up to the RWST.		X	
Recovery of RCP Seal Cooling	In some plants there are means of alternate cooling for RCP seals that could be relied on in scenarios involving loss of CCW. However, the alignment of the system is manual and requires operator action.	X		X
ATWS Response	Upon failure of RPS, the operator should perform several actions, starting with manual scram, ensuring turbine trip, and most importantly initiating boron injection.		X	X
Isolation of ISLOCA	In some plants there is a capability to isolate an interfacing systems LOCA through manual actions. Operator failure to isolate on interfacing LOCA in the LPI system is considered risk significant in these plants.	X		X
Initiation of AFWS	This human action involves failure to manually start the motor driven AFW pump, given auto start failure, and failure to manually start locked out turbine driven AFW pump.		X	

<u>Notes for Table 2, RIM No. 2 PWRs</u>:

3. The Systems listed as *Risk Important at Most Plants* were derived from NUREG-1560, “Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance”, listed as: important for most PWRs in Table 3.9; or related to those containment performance perspectives of Table 4.9 listed as having significant probability for most BWRs.

Table 2, Risk Information Matrix No. 2: PWRs (continued)

4. Contains those systems generically determined to have high Risk Achievement Worth (RAW $\succ$ 10), that were not contained in the first part of this Table.

5. Table developed to provide generic PWR risk insights for use in the development of a generic Risk-informed baseline inspection program.

6. PWR Systems that Were *Potentially Important* were extracted from the IPE Database, when they appeared in the top ten sequences of several plants. PWR Human actions that Were *Potentially Important* were extracted from NUREG-1560 Table 5.3, when they occurred in more than two plants (but less then 50% of the plants).

Table 2, Risk Information Matrix No. 2: BWRs

BWR Systems that Were *Risk Important at Most Plants* and Reasons for Importance
<u>Not prioritized within this portion of the Table</u>

Important Systems From IPEs	Reasons for Importance	Cornerstones		
		Initiating Events	Mitigating Systems	Barrier
Systems selected based on CDF contribution from NUREG-1560				
Offsite AC Power	The loss of offsite power is an important initiator event, if lasting beyond 30 minutes. Loss of offsite power could occur as a result of adverse weather conditions, switch yard activities, degraded grid, component failure, design inadequacy, or human error (e.g., when maintenance on part of system is being performed). Losses of offsite power due to adverse weather conditions could be important since the harsh weather condition such as freezing could impact other mitigating systems. AC system design and maintenance contribute to the overall reliability of the system (including both offsite and onsite AC power). Features important include: Number of offsite lines, design of switch yard, cross-tie capability, availability and reliability of equipment. Some losses of offsite power could be recovered by operator actions. Recovery of offsite power is shown to be important in PRAs.	X	X	
Onsite Emergency AC Power	Common cause failures of multiple emergency diesel generators has been a major risk concern. The unavailability of one diesel generator either due to failure or maintenance, combined with the failure of the other safety trains that are fed from the remaining emergency bus, have also been a major contributor to risk. Therefore, configuration control would be important during maintenance or emergent work on EDGs. Recovery of Emergency AC or EDGs in SBO sequences is important.		X	

Table 2, Risk Information Matrix No. 2: BWRs (continued)

BWR Systems that Were *Risk Important at Most Plants* and Reasons for Importance **Not prioritized within this portion of the Table**				
Important Systems From IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Systems selected based on CDF contribution from NUREG-1560				
HPCI	Used to provide injection and remove Decay Heat (DH) from the core on an SBO. Availability and Redundancy of High Pressure Injection Systems (HPCI & RCIC) is important to Transients with Loss of Injection Sequences On a transient with loss of DHR, two issues are: NPSH Problems with ECCS in the Suppression Pool; and the Capability of the ECCS to Pump Saturated Water. Mitigating system redundancy can be impacted by harsh environments in containment (before cont. failure). On a transient with loss of DHR, resulting in containment failure, Mitigating system redundancy (for Systems Located Outside of Containment and Rx Bldg) can be impacted as a result of harsh environments in adjacent structures (after cont. failure).		X	
RCIC	Used to provide injection and remove Decay Heat (DH) from the core on an SBO. Availability and Redundancy of High Pressure Injection Systems (HPCI & RCIC) is important to Transients with Loss of Injection Sequences. On a transient with loss of DHR, two issues are NPSH Problems with ECCS in the Suppression Pool; and the Capability of the ECCS to Pump Saturated Water. Mitigating system redundancy can be impacted by harsh environments in containment (before cont. failure). On a transient with loss of Decay Heat removal (DHR), resulting in containment failure, Mitigating system redundancy (for Systems Located Outside of Containment and Rx Bldg) can be impacted as a result of harsh environments in adjacent structures (after containment failure).		X	
HVAC for HPCI & RCIC	Loss of HVAC may cause common mode failure of HPCI & RCIC.		X	

Table 2, Risk Information Matrix No. 2: BWRs (continued)

BWR Systems that Were *Risk Important at Most Plants* and Reasons for Importance Not prioritized within this portion of the Table				
Important Systems From IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Systems selected based on CDF contribution from NUREG-1560				
Diesel-Driven Firewater	Used to provide injection and remove DH from the core on an SBO. Use of firewater requires significant planning and training, but is credited in several plants.		X	
Isolation Condenser (IC)	Used to remove Decay Heat (DH) from the core on an SBO. A larger IC capacity provides more time before makeup to IC is required and allows more time for operators to recover AC power. The ability of the IC to remove DH is defeated if an RV (Relief Valve) sticks open.		X	
DC Batteries	DC power is required for operation of the AC independent systems (e.g., IC, HPCI, RCIC, and ADS or SRVs). Battery depletion times typically range from 2 to 14 hours. Load shedding is important to extend depletion time. This provides more time for recovery of AC power. Failures of support equipment, such as DC power can fail the EDGs.		X	
DC Buses	Loss of Dc buses is an important initiator since it can cause reactor trip and compromise the operation of the mitigating systems.	X	X	
Service Water	Failures of support equipment, such as SW can fail the EDGs. Failure of EDGs due to loss of SW is an important contributor to SBO. SW design is very unit specific.		X	

Table 2, Risk Information Matrix No. 2: BWRs (continued)

BWR Systems that Were *Risk Important at Most Plants* and Reasons for Importance Not prioritized within this portion of the Table				
Important Systems From IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Systems selected based on CDF contribution from NUREG-1560				
Automatic Depressurization System and Relief Valves (ADS/RVs)	The ability of the IC to remove DH (e.g., on an SBO) is defeated if a RV sticks open. Failure to Depressurize is important on Transients with Loss of Injection Sequences. Many plant procedures direct operators to inhibit ADS on transients. When high pressure injection fails, operators must manually depressurize with ADS. On selected sequences such as SBO, depressurization is required after failure of high pressure injection systems to allow for injection with low pressure systems. A complicating factor is that some procedures initially direct the operator to inhibit ADS. In some PRAs this appears in cutsets up to 45 % of CDF. Excessive SRV discharge to a hot suppression pool is found to lead to late containment failure.		X	
MSIVs	Inadvertent MSIV closure is an important initiator in BWRs. Failure of MSIV to close in steam line break scenarios could result in LOCA outside containment.	X		
Common Support Systems for Injection and DHR Systems	Injection System and DHR system dependencies on Support Systems Defeats Redundancy. Support system (cooling water, Inst. Air, and AC or DC power) failures can impact multiple mitigating systems.		X	

Table 2, Risk Information Matrix No. 2: BWRs (continued)

BWR Systems that Were *Risk Important at Most Plants* and Reasons for Importance **Not prioritized within this portion of the Table**				
Important Systems From IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Systems selected based on CDF contribution from NUREG-1560				
Decay Heat Removal (DHR)	An important class of sequences is Transients with Loss of DHR. In these sequences, coolant injection succeeds but DHR fails and RVs open, sending steam to SP. Many licensees made Mods to DHR systems and improvements to ensure continued system operation under harsh environment conditions. Some PRAs had Limited Analysis to Support DHR Success Criteria; No Credit was taken in Some Plants for Alternate DHR Systems (e.g., Venting). Some plants had higher contribution from these transients due to not crediting all alternate DHR systems. Also, not all plants treated the results of harsh environments the same. On a transient with loss of DHR, two issues are NPSH Problems with ECCS in the Suppression Pool; and the Capability of the ECCS to Pump Saturated Water. Mitigating system redundancy can be impacted by harsh environments in containment (before cont. failure). On a transient with loss of DHR, resulting in containment failure, Mitigating system redundancy (for Systems Located Outside of Containment and Rx Bldg) can be impacted as a result of harsh environments in adjacent structures (after containment failure).	X	X	
Containment Heat Removal (CHR) systems/ Drywell Spray	Failure of CHR systems is important to Transients with Loss of DHR sequences. Less Restrictive Drywell Spray Initiation Criteria are important since high pressure loads at time of core debris melt- through of Reactor vessel and Fuel Coolant Interaction are important contributors to early containment failure.		X	X
Containment Venting	Operator Training on Depressurization of Containment is important since high pressure loads at time of core debris melt- through of RV & FCI are important contributors to early cont. failure. Venting is found to be an effective means of avoiding uncontrolled containment failure for Mark I and III containments in core damage events.		X	X

Table 2, Risk Information Matrix No. 2: BWRs (continued)

BWR Systems that Were *Risk Important at Most Plants* and Reasons for Importance Not prioritized within this portion of the Table				
Important Systems From IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Systems selected based on CDF contribution from NUREG-1560				
Feedwater	Loss of feed water is an important initiator in some BWRs. Availability and Redundancy of High Pressure Injection Systems is important to Transients with Loss of Injection Sequences	X		
Suppression Pool (SP) Cooling	On a transient or LOCA sequence, with failure of the PCS and the SRVs open, containment temperature and pressure increase and must be controlled. This can be done by containment heat removal, suppression pool cooling, or containment venting. Actions are required to remove DH before adverse conditions are reached (e.g., hi SP temperature leading to loss of ECCS pumps).		X	X
Alternative Water Sources for Flooding of Drywell Floor	Ensuring the Drywell Floor is Flooded in Core Damage Event is important to prevent late containment failure since High pressure and temperature loads caused by CCI are an important failure mode.			X
Combustible Gas Control/Igniters in Containment of Mark III BWRs	Hydrogen burns are an important contributor to early containment failures Mark III IPEs. Combustible gas burns are important in Mark IIIs to prevent late containment failure.			X

Table 2, Risk Information Matrix No. 2: BWRs (continued)

<table>
<tr><th colspan="5">Additional BWR Items that Were Risk Important at Most Plants and Reasons for Importance</th></tr>
<tr><th rowspan="2">Important Systems from IPEs</th><th rowspan="2">Reasons for Importance</th><th colspan="3">Cornerstones</th></tr>
<tr><th>Initiating Events</th><th>Mitigating Systems</th><th>Barrier</th></tr>
<tr><th colspan="5">Additional Systems - Important based on high RAW (note 2)</th></tr>
<tr><td>RPS/CRDMs</td><td>Failure of the RPS could result in ATWS scenarios which is important risk contributor in some BWRs.</td><td></td><td>X</td><td></td></tr>
<tr><td>4160 V switchgear</td><td>Failure or unavailability of 4160 V switchgear could result in loss of offsite power and compromise the operation of mitigating systems.</td><td>X</td><td>X</td><td></td></tr>
<tr><th colspan="5">Shutdown Risk Insights</th></tr>
<tr><th colspan="5">BWR shutdown risk insights to be determined later.</th></tr>
<tr><th colspan="5">Fire Protection Insights</th></tr>
<tr><td colspan="2">Fires in BWRs are relatively important to Total CDF. Fires in the following areas are typically important: control room, cable spreading room, and various switchgear rooms. The actual fire areas or rooms that are important are plant specific and are identified in the plant IPEEEs or PRAs.</td><td>X</td><td>X</td><td></td></tr>
<tr><td colspan="2">The three general areas to address fire risk are: prevention, mitigation/suppression, and safe shutdown. NUREG/CR-4230 found that the following fire protection system features were the most important to risk: qualified cabling, 3-hour fire barriers, automatic suppression systems, and automatic doors & dampers.</td><td></td><td>X</td><td></td></tr>
<tr><th colspan="5">Internal Flooding Insights</th></tr>
<tr><td colspan="2">Internal flooding sequences are not dominant contributors to CDF in most BWRs. These events involve rupture of water lines or operator errors that result in the release of water that can cause the failure of required mitigating systems through: loss of cooling, submergence or spraying. The most important factor is the plant-specific layout. Separation and compartmentalization reduce the impact of internal flooding. For BWRs, no floods were identified that could fail all necessary mitigating systems.</td><td>X</td><td>X</td><td></td></tr>
<tr><td colspan="2">A few BWR plants identified important sequences that generally involve Service Water System breaks that impact equipment both through loss of SW cooling and flood-related impacts on other mitigating systems (such as electrical switchgear).</td><td>X</td><td>X</td><td></td></tr>
</table>

Table 2, Risk Information Matrix No. 2: BWRs (continued)

Additional BWR Items that Were *Risk Important at Most Plants* and Reasons for Importance				
Important Systems from IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Internal Flooding Insights (continued)				
Risk significant improvements at some plants were: periodic inspection of susceptible piping to reduce potential of flooding, enhancement of flood response procedures, and training for flooding, including isolation of flood source.		X	X	
Human Actions				
Perform Manual Depressurization	On selected sequences, depressurization is required after failure of high pressure injection systems to allow for injection with low pressure systems. A complicating factor is that some procedures initially direct the operator to inhibit ADS. In some PRAs this appears in cutsets up to 45 % of CDF.		X	
Containment Venting Align Containment or Suppression Pool Cooling	On a transient or LOCA sequence, with failure of the PCS and the SRVs open, containment temperature and pressure increase and must be controlled. This can be done by containment heat removal, suppression pool cooling, or containment venting. Actions are required to remove DH before adverse conditions are reached (e.g., hi SP temperature leading to loss of ECCS pumps).		X	X
Initiate SLC	Manual initiation of Standby Liquid Control (SLC) system is needed ATWS Scenarios.		X	

Table 2, Risk Information Matrix No. 2: BWRs (continued)

BWR Items that Were *Potentially Important* and Reasons for Importance Not prioritized within this portion of the Table				
Important Items From IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Systems selected based on inclusion in several plants in the IPE Data Base				
Loss of instrument air (LOIA)	LOIA was important as an initiating event in a number of IPEs.	X		
LPI	The high redundancy and diversity of coolant injection systems in BWRs lowers the importance of LOCAs. However, they do still appear as contributors to the total CDF. The LPI System is an important mitigator for large and medium LOCAs. One key aspect of the LPI system is the proper functioning of the low reactor vessel pressure permissive.		X	
Core Spray (CS)	The high redundancy and diversity of coolant injection systems in BWRs lowers the importance of LOCAs. However, they do still appear as contributors to the total CDF. CS is used as one of the injection systems for LOCAs.		X	
Standby Liquid Control	Important for the injection of boron during ATWS events.		X	
Alternate Rod Insertion (ARI)	Failure of the ARI system appears as contributor to ATWS sequences in many plants.		X	
Human Actions				
Level Control in ATWS	Effective Rx Vessel level control is needed during an ATWS in order to reduce core power.		X	X
Align/Initiate Alternative Injection	This relates to loss of injection and loss of DHR sequences. Alternate sources of injection include: SW, firewater, CRD, FW booster pumps, SP cleanup, and a few plant unique systems.		X	

Table 2, Risk Information Matrix No. 2: BWRs (continued)

BWR Items that Were *Potentially Important* and Reasons for Importance **Not prioritized within this portion of the Table**				
Important Items From IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Human Actions (continued)				
Recover Ultimate Heat Sink	The importance of recovery of SW or the ultimate heat sink depends on the cooling requirements of mitigating systems and the time available before they fail after loss of cooling. Recovery is also needed to allow adequate removal of DH from the core. Some of these are possible just from the main CR, while others require local operator actions.		X	
Inhibit ADS	Some IPEs conclude that core damage will occur if ADS is not inhibited in an ATWS event due to instabilities created at low pressures.		X	X
Mis-calibrate Pressure Switches	Various pressure switches are important for initiating ECCS and operating ECCS permissives. Common cause mis-calibration of these switches can affect multiple trains of safety systems.		X	
Initiate Isolation Condenser	For the early BWR plants, this action is important during accidents to ensure the continued viability of the cooling from the IC.		X	
Control FW Events (e.g., After Loss-of-Instrument Air)	The actions of operators to properly control the FW system can be important in transient and small LOCA sequences.	X	X	
Manually Initiate Core Spray or Other Low-Pressure System	Where these low pressure injection systems fail to automatically actuate, then operator action to manually initiate them becomes necessary.		X	

Table 2, Risk Information Matrix No. 2: BWRs (continued)

BWR Items that Were *Potentially Important* and Reasons for Importance **Not prioritized within this portion of the Table**				
Important Items From IPEs	**Reasons for Importance**	**Cornerstones**		
		Initiating Events	**Mitigating Systems**	**Barrier**
Human Actions (continued)				
Mis-calibrate Low-Pressure Core Spray Permissive	This is mis-calibration of the permissives needed to open the low pressure core spray and LPCI injection valves, which are needed in several sequences. Also included is the failure to restore these permissives after testing.		X	
Provide Alternate Room Cooling (In Event of Loss of HVAC)	On transient sequences, loss of HVAC (due to various reasons) can jeopardize ECCS equipment operation. The operators may be able to take actions to provide alternate room cooling, such as opening doors and providing blowers.		X	
Recover Injection Systems	This action relates to operator recovery of failed or unavailable injection systems and can be important in sequences where such failures are dominant.		X	
DC Load Shedding After SBO	While not well modeled, the shedding of DC loads is needed to extend the battery charge in order to operate the AC independent HPCI and RCIC systems and to keep the SRVs open (to allow low pressure vessel injection from a diesel-driven fire pump). This extends the time to core damage and that operators have for recovery of AC.		X	

Notes for Table 2, RIM No. 2, BWRs

1. The Systems listed as *Risk Important at Most Plants* were derived from NUREG-1560,"Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," listed as: important for most BWRs in Table 3.2; or related to those containment performance perspectives of Table 4.2 listed as having significant probability for most BWRs.

1. The Systems listed as *Risk Important at Most Plants* were derived from NUREG-1560,"Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," listed as: important for most BWRs in Table 3.2; or related to those containment performance perspectives of Table 4.2 listed as having significant probability for most BWRs.

2. Contains those systems generically determined to have high Risk Achievement Worth (RAW > 10), that were not contained in the first part of this Table.

3. BWR Systems that Were *Potentially Important* were extracted from the IPE Database, when they appeared in the top ten sequences of several plants. BWR Human actions that Were *Potentially Important* were extracted from NUREG-1560 Table 5.1, when they were identified as important in more than two plants (but less then 50% of the plants).

4. Table developed to provide generic BWR risk insights for use in the development of a generic Risk-informed baseline inspection program.

ASSESSMENT PROCESS

Team Leader	Michael Johnson
Team Members	John Flack Tim Frye David Gamberoni Bob Haag Don Hickman Bill Jones Alan Madison Glenn Meyer Chris Miller Bob Pascarelli Bill Reckley Mark Satorious

Attachment 4

EXECUTIVE SUMMARY

A multi-disciplined, interoffice task group was formed to continue developing assessment process improvements that were initiated under the Integrated Review of Assessment Processes (IRAP) effort. The efforts described in this attachment focus on the design of a new performance assessment process within the structure of regulatory oversight described in Attachment 2 to this Commission paper. The assessment task group included representatives from each regional office and the Offices of Nuclear Reactor Regulation (NRR), Analysis and Evaluation of Operational Data (AEOD), Enforcement (OE), and Nuclear Regulatory Research (RES).

The charter of the assessment task group was to develop a process that will allow the NRC to integrate various information sources relevant to licensee safety performance, make objective conclusions regarding their significance, take actions based on these conclusions in a predictable manner, and effectively communicate these results to the licensees and to the public. The efforts of the assessment task group were closely coordinated with the framework, inspection, and enforcement efforts, which are also described in this Commission paper.

Several key principles were identified that have a direct effect on the assessment process design:

- Both performance indicators (PIs) and inspection results will be inputs to the assessment process

- Performance indicators and cornerstone inspection areas (inspection results grouped by cornerstone area) will have established thresholds

- Crossing PI or cornerstone inspection area thresholds will have similar meaning and will result in the NRC considering a similar range of actions

A review system, shown in Table 3.1, was developed that provides continuous, quarterly, mid-cycle, and end-of-cycle (annual) reviews of licensee performance data (PIs and inspection results). The system is designed so that the lower level reviews are informal reviews of performance data and are not resource intensive. The mid-cycle review is more formal and is focused on assessing performance to determine appropriate NRC inspection actions. The mid-cycle review generates an inspection planning letter. The end-of-cycle review generates both an assessment report and an inspection planning letter. The agency action review is reserved for plants requiring consideration of agency-wide actions. This review is analogous to the current Senior Management Meeting (SMM); however, the focus has been changed from an assessment activity to an oversight and agency-level action approval function.

An action matrix, shown in Table 3.2, was developed to provide guidance for consistent consideration of actions. The actions are graded across five ranges of licensee performance in all response categories (management meeting, licensee action, NRC inspection, and regulatory actions) and in terms of annual communication of assessment results. Action decisions are triggered directly from the threshold assessments of PIs and inspection areas.

For example, a single PI or cornerstone inspection area crossing its threshold would require the NRC to <u>consider</u> the actions listed in the second performance range of the action matrix, such as inspection follow-up to determine the cause of the assessment input degradation. More significant changes in performance, such as one degraded cornerstone would result in the consideration of more significant actions.

The action matrix is not intended to provide guidance that is excessively rigid. It establishes expectations for interactions, licensee actions, and NRC actions. It does not preclude the NRC from taking less action or some additional action, when justified. The key point is that assessment results are not altered; they are the PI and cornerstone inspection area results. The action decisions can be modified, when appropriate.

The communication of assessment results involves quarterly updates of assessment data, semiannual inspection planning letters, and annual assessment reports. The Commission has negative consent approval of all assessment results and NRC actions prior to an annual Commission meeting. <u>All</u> assessment results are released at the Commission meeting to provide proper balance and context. Following the Commission meeting, public meetings with individual licensees will be held to discuss assessment results. This differs from the current SMM, which focuses primarily on poor performers.

CONTENTS

Tables

iii

1 INTRODUCTION

1.1. Background

The individual components of the current Nuclear Regulatory Commission's (NRC) assessment processes for operating commercial nuclear reactors were developed and implemented at different times. The systematic assessment of licensee performance (SALP), was being developed before the Three Mile Island accident and was implemented in 1980. It was intended to provide a systematic, long-term, integrated evaluation of overall licensee performance.

The senior management meeting (SMM), was developed in response to the 1985 Davis-Besse loss-of-feedwater event and was first implemented in 1986. It was developed to bring to the attention of the highest levels of NRC management those plants where operational safety performance was of most concern.

Plant performance reviews (PPRs) were developed to provide for better allocation of NRC resources and were implemented in 1988. PPRs are conducted more frequently than SALPs or SMMs and were developed to provide mid-course adjustments in inspection focus in response to changes in licensee performance and emerging plant issues.

The plant issues matrix (PIM) provides an index of the primary issues, generated through inspection findings and licensee event reports (LERs), that are evaluated during the SALP, SMM, and PPR processes. It was developed as part of the effort to improve the integration of inspection findings following the South Texas Lessons Learned Task Force, and was implemented in 1996.

Each process served a specific purpose and the individual processes were improved incrementally since implementation. However, the manner in which the NRC assesses the safety performance of its licensed nuclear utilities has become a matter of concern both within the NRC itself and by industry and stakeholders. The following weaknesses were identified:

- Many of the process components are redundant and have similar end products.
- The assessment criteria differ between process components, especially SALP and the SMM, and are not viewed as sufficiently objective.
- The processes are subject to inconsistent implementation among the regions.
- The processes are more resource-intensive than originally intended, particularly when the safety-significance of the results obtained is considered.
- The results of the current processes (especially SALP scores) are misunderstood with respect to safety.

In summer 1997, the Commission approved the staff's request to perform an integrated review of the processes used to assess licensee performance. This approval and previous staff requirements memoranda provided several issues to be addressed in this review, including that any assessment process developed as a result of staff efforts:

- Have clear roles and responsibilities (including for the Commission)
- Maintain data integrity (which implies that the process does not distort the data and that

there is precision in the data; data may be weighted, but weightings must be known and consistent)

2. Include a decision model/criteria
3. Identify risk significance
4. Be simple/non-redundant/efficient
- Be complete (which implies that all relevant information is included)
- Be well integrated (that is, both the process itself is well integrated, and information inputs and output are integrated)
5. Include a self-assessment process (to evaluate its own effectiveness)
6. Be objective (but not necessarily quantitative)
7. Be consistent
8. Be timely
9. Be validated
- Include performance ratings (for both good and poor performance)
10. Be predictive (and reduce reliance on events)
- Be scrutable (be open, clear, and transparent; include public involvement)

Initial efforts to perform the integrated review commenced in September 1997, with the Integrated Review of Assessment Processes (IRAP). Preliminary results from the IRAP were forwarded to the Commission in March 1998 via SECY-98-045, "Status of the Integrated Review of the NRC Assessment Process for Operating Commercial Nuclear Reactors." These results are summarized below in section 1.2. While the NRC was performing the IRAP, the Nuclear Energy Institute (NEI) developed an alternative proposal for regulatory oversight (including assessment). The Commission subsequently approved the staff soliciting public comments on both the IRAP preliminary results and on other potential changes to the assessment processes.

Ongoing public interactions between the NRC staff, the Advisory Committee on Reactor Safety, the Commission, and industry caused the staff to broaden its review of assessment to oversight in general (inspection, assessment, and enforcement). This resulted in a change in direction from that proposed by IRAP to one within a new regulatory structure that is described in Attachment 2 of this Commission paper that includes "cornerstones" of safety performance.

The new approach resulted from the confluence of the two proposals generated by the NRC staff and industry (NEI), respectively, aimed at addressing the problems with the current processes. The assessment process proposed by the NRC staff, called the integrated assessment process, would have used inspection findings as its primary data source and would have provided a mechanism for checking the inspection-based assessment results against other data sources, such as industry performance indicators (PIs), the trending methodology developed by the NRC's Office for the Analysis and Evaluation of Operational Data (AEOD), and licensee-generated self-assessment data. In contrast, the NEI proposal would have used a performance-based model that relied primarily on licensee-generated PI data and required minimal NRC involvement unless a performance threshold was crossed. Each of these proposals is briefly described below.

Initial development of the new regulatory oversight structure occurred during a public workshop that was held on September 28 through October 1, 1998. The results of that workshop were a general acceptance of the framework concept, modifications and additions to the approach, and high-level development of the framework.

The following subsections briefly describe the integrated assessment process, the NEI proposal, and the new assessment process. Section 3 of this attachment describes the processes by which the assessment will be conducted.

1.2. NRC Integrated Assessment Process

The IRAP involved NRC regional and headquarters (HQ) staff, and used a principle-based approach (i.e., starting with objectives and attributes, then designing processes to achieve them) to evaluate existing processes and design new ones, where necessary. The IRAP proposed that the inspection program would continue to observe licensee performance and document those observations in inspection reports. Performance issues would be entered into the PIM and would be assigned a significance rating and template category tag. The template would be a tool for sorting inspection issues and include both functional and cross-functional categories. Functional categories include: operational performance, material condition, engineering/design, and plant support. Cross-functional categories include: human performance, problem identification and resolution, and programs and process. Each PIM entry would be binned into both a functional and cross-functional category.

The graded PIM entries would be aggregated by template category, and numerical thresholds would be used to produce an assessment rating for each template category.

The performance of every plant would be assessed annually, at a regional meeting. This meeting would allow for the review of, and reconciliation between, the template assessment and other indicators. A decision logic model would be applied to the assessment results to determine the range of NRC actions that should be considered as well as the appropriate communication methods. NRC actions would be taken in a graded approach, with different levels of NRC management responsible for the action, depending upon licensee performance. Assessment results would be issued in writing to both the licensee and the public and would be reviewed with the licensee at a public meeting; again, NRC (and licensee) participation in the public meeting would be graded based upon the assessment results.

This proposal focused on actions and proposed elimination of labels such as watchlist and superior performer plants.

1.3. NEI Risk-Informed, Performance-Based Assessment Process

The NEI approach would use the existing regulatory requirements (primarily Title 10 of the Code of Federal Regulations [CFR] Parts 50 and 100) as a basis for setting licensee performance expectations that relate to public health and safety. For assessment purposes, these performance expectations would be grouped into three tiers:

- Tier I: Public health and safety -- maintaining the barriers for radionuclide release, and controlling radiation exposure and radioactive materials.

- Tier II: Safety performance margin -- minimizing operational events that could challenge the barriers and ensuring that engineered safety systems can perform their intended safety functions.

- Tier III: Overall plant performance -- plant safety performance trends are used as leading indicators for problems that might develop in the Tier II performance areas.

Each performance expectation would have a set of PIs that would be used to evaluate the achievement of the expectation. For Tier I, the performance expectation of barrier integrity would be evaluated using three PIs: reactor coolant system (RCS) activity (level of fission products), RCS boundary (leakage rate from primary boundary), and containment integrity. The performance expectation of control of exposure and radioactive materials would also have three PIs: emergency preparedness, radioactive material control (release and shipment of radioactive materials), and exposure control (for both workers and the public).

For Tier II, the performance expectation related to operating challenges would be monitored using four PIs: unplanned automatic scrams, safety system actuation, shutdown operating margins, and unplanned operating transients. The performance expectation related to mitigation capability would be assessed on the basis of high risk-significant structures, systems, and components performance.

Tier III would be monitored using an index of plant safety, which would be trended to show the direction overall safety performance could be headed.

With the exception of the Tier III trending indicators, all indicators would have an objective regulator threshold and a safety threshold value. The regulator threshold defines the level of performance at which the safety performance margin has declined to a point where regulatory attention may be warranted. The safety threshold defines the level of performance at which the safety performance margin has declined to a point where plant operation is not permitted until corrective action is taken to restore margin.

The thresholds, in turn, define three response bands: a utility response band, a regulator response band, and an unacceptable band. If performance is within the utility response band, utility management would maintain performance within the control band; the NRC would perform baseline inspections or opt to evaluate/participate in licensee self-assessments and audits, particularly in those areas not covered by the safety performance indicators, and monitor the PIs.

The regulator response band defines the point at which the regulatory response increases to questioning the adequacy of licensee corrective actions and programs and processes related to the performance area for which the band has been crossed. The degree of regulator response would be determined by how close performance is to the unacceptable band. Performance far from the unacceptable band threshold would receive minimal regulatory action while performance closer to the unacceptable band would receive more aggressive action, such as increased inspection, confirmatory action letters (CALs), and civil penalties (CPs).

The unacceptable band defines the point at which plant operation is no longer allowed until corrective action is taken.

The licensees and the NRC would have different, but complementary, roles in this assessment approach. The NRC would first assess results, by verifying the PIs and reviewing inspections and corrective actions. Based on the assessment results, the NRC would develop and implement inspection plans with the scope of those plans being defined by the response bands, as described above. Similarly, the NRC would take regulatory action as indicated by the response band. The licensee would monitor and report on the PIs; inform the NRC of its self-assessment and audit plans and make the results of self-assessments and audits available to the NRC prior to planned inspections; and perform root cause analysis, identify corrective action, and report the status of corrective actions to the NRC prior to NRC regulatory actions.

1.4. New Performance Assessment Process

The new regulatory oversight framework is a hierarchical structure that begins with a focus on the NRC's overall safety mission and identifies strategic areas in which performance must be maintained for the overall safety mission to be achieved. Each strategic performance area, in turn, has a set of cornerstones (areas) that support the strategic performance area. The cornerstones comprise those major essential elements, the presence of which provide reasonable confidence that licensee performance is such that goals are achieved. Performance must be maintained in the cornerstone areas to achieve the agency's strategic performance area goals and to meet the overall safety mission. It is important to note that other regulatory processes, such as licensing activities, also contribute toward meeting these goals.

PIs and inspection provide the data to assess performance. Decision thresholds are used to determine the regulatory action warranted by licensee performance. The new regulatory oversight framework and its development are discussed in detail in Attachment 2 to this Commission paper.

Within the framework, each cornerstone has an underlying structure comprising its desired results, attributes important to achieving those results, areas to measure, and means of measurement. Several characteristics of that structure are important to the subsequent discussions.

First, the measurement methods for each cornerstone comprise a mix of PI data and inspection results grouped by cornerstone inspection area. While there was a preference for identifying PIs for each cornerstone, it was recognized that there are gaps in the information provided by objective PI data. Therefore, complete assurance of licensee performance in a particular cornerstone area will require examining *both* PI and inspection data.

In addition, in much the same way as was proposed by NEI, it is expected that each PI and cornerstone inspection area will have thresholds or bands which determine when regulator response is triggered.

Finally, it is important to note that the workshop resulted in a set of defining principles or boundary conditions to guide more detailed development of the framework and the processes for implementing it. These defining principles include:

- There will be a risk-informed baseline inspection program (RIBIP) that establishes the minimum regulatory interaction for all licensees
    - RIBIP will cover those risk-significant attributes of licensee performance not adequately covered by PIs
    1. RIBIP will also verify the accuracy of the PIs and provide for event response

11. PIs supplemented with some inspection will form the basis for licensee assessment

- Thresholds can be set for licensee safety performance, where increased NRC interaction (including enforcement) would be warranted

- Enforcement actions taken (e.g., number of cited violations, the amount of CP) should not be an input into the assessment process. However, the issue that resulted in the enforcement action will continue to be an input to assessment

- Assessment process results might be used to modulate enforcement actions (although assessment results would not affect the assessment of severity of the violation)

1.5. Follow-On Efforts

The NRC staff formed four task activities: the technical framework group, the inspection group, the assessment group, and enforcement. This attachment documents the results of the assessment task group. Task force members comprised a broad spectrum of NRC staff with expertise in assessment, including representatives from the Office of Nuclear Reactor Regulation (NRR), Office for Analysis and Evaluation of Operational Data (AEOD), Office of Nuclear Regulatory Research (RES), Office of Enforcement (OE), Office of the Executive Director for Operations (OEDO), and the regions. In addition, numerous regularly scheduled public working meetings were held to solicit timely feedback on the work of the NRC staff. A public meeting that focused specifically on the assessment team's efforts was held on November 12, 1998. Conclusions of that meeting have been incorporated into this attachment.

Key interfaces with the other task groups are discussed below.

1.5.1. Technical framework group.   The technical framework group was charged with completing the development of the new regulatory oversight structure, including identifying cornerstones, and specifying in detail all PIs and inspection bases needed to provide a representative sample of performance in each cornerstone, as well as thresholds for judging each piece of assessment data.

The assessment group communicated several key assumptions to the technical framework group, including that:

- to the extent possible, multiple use of PIs within the framework should be minimized

- the meaning of the thresholds for all PIs and cornerstone inspection areas should be similar -- that is, it will take an approximately equivalent change in risk or performance to cross into the regulator response threshold

1.5.2. Inspection group. The inspection group (as discussed in Attachment 3 to this Commission paper) was charged with defining the scope of the RIBIP.

Methods will be developed such that each inspection finding entered into the plant issues matrix (the inspection finding database) is tagged with a code indicating to which cornerstone inspection area or PI it applies (depending on whether it is primary data source being used to fill a gap in the cornerstone sampling data or whether it is being used to verify a PI), its risk significance, and context for the finding.

1.6. The Assessment Task Group

The purpose of the assessment task force was to develop the process that will allow the NRC to integrate various information sources relevant to licensee safety performance, make objective conclusions regarding their significance, take actions based on these conclusions in a predictable manner, and effectively communicate these results to the licensees and to the public. This task group also developed recommendations on the best methods to transition from the existing regulatory oversight processes and implement the proposed new oversight process.

The scope of this effort was limited to operating power reactors, and does not include, for example, permanently shutdowns or decommissioned reactors or fuel fabrication or material licensees.

<u>Tasks</u>

Key tasks for the assessment task group included:

- Developing a methodology for the integration of information inputs (PI results as well as risk-significant inspection findings and other information sources) so that the assessment results are objective

- Developing decision criteria or a decision model so that NRC actions can be taken in a manner that is scrutable and predictable by both the licensees and the public

- Identifying the necessary program requirements to support a voluntary licensee assessment data reporting process and developing a recommendation on how those licensees who do not participate in a voluntary program would be accounted for and assessed

- Determining methods for communicating to both licensees and the public the assessment results and NRC actions taken; evaluating the appropriateness of taking a graded approach to communications

- Determining how licensee performance in response to NRC actions is monitored and measured and how the results feed back into the assessment process

- Developing the assessment process mechanisms, including determining the appropriate frequency for routine assessment of licensee performance; identifying the appropriate staff positions for conducting the assessment and determining their responsibilities; determining the level of senior NRC management involvement required for the performance assessments; and developing a process for handling changes in the assessment input[5] results as they occur so that appropriate action is not reliant on the performance of a periodic assessment

- Developing a methodology for the continuous self-assessment of program effectiveness subsequent to implementation

- Developing a recommendation for implementing the new process, including the necessary actions required to validate the new process prior to implementation and whether implementation should occur via a phased-in approach for all licensees or a pilot program with targeted or voluntary participation

- Determining appropriate methods to transition from the current assessment processes to the new approach, including providing a recommendation regarding the continued suspension of SALP

- Interfacing with OE to develop concepts of how assessment results affect enforcement actions and how "regulatory significance" is defined and used in the oversight process

Approach

---

[5] "Assessment input" is used as a generic term which encompasses all data used in assuring performance within a cornerstone, and includes both PIs and cornerstone inspection areas.

Because the IRAP effort began by developing a high-level assessment process, its results provide a generic framework for assessment. Briefly, the process steps suggested by IRAP are arranged into three sub-processes and include:

1. process development and evaluation, comprising those activities related to 1) developing a decision model by which assessments are performed, 2) providing guidance regarding information collection needs, and 3) conducting an evaluation of the assessment process to effect process improvements; the first two of these are intended to only occur once in their full elaboration

2. core assessment, which comprises 1) conducting the assessment and deciding on a course of action using the decision model, and 2) communicating and implementing the necessary course of action, and

3. on-going evaluation, which is an on-going checking of the reasonableness of the assessment results and the decisions made as a consequence of those results.

Thus the assessment team adopted the IRAP process representation as a starting point for its work, recognizing that many of the details may not apply to the new regulatory oversight framework.

This attachment focuses on the core assessment sub-process and the process evaluation portion of the process development and evaluation sub-process. The decision model and implementation guidance are discussed in the context of where and how they are applied in the process. Because PIs and inspection findings are used together in the assessment approach, the group determined that an elaborate evaluation that brings in other data to contradict assessment results is not needed.

2 DEFINING PRINCIPLES AND DEFINITIONS

By the conclusion of the workshop, thoughts about the new regulatory oversight framework had progressed in directions that challenged some of the defining principles stated initially. Further work on the framework, inspection program, and assessment process provided additional clarification on the defining principles. A modified set of defining principles, which reflect the conceptual changes to the assessment process, are provided below.

Defining principles that remained unchanged following the workshop include:

- There will be a risk-informed baseline inspection (RIBIP) program that establishes the minimum regulatory interaction for all licensees
  - RIBIP will cover those risk-significant attributes of licensee performance not adequately covered by PIs
  4. RIBIP will also verify the accuracy of the PIs and provide for event response

- Thresholds can be set for licensee safety performance, where increased NRC interaction (including enforcement) would be warranted

- Enforcement actions taken (e.g., number of cited violations, the amount of CP) should not be an input into the assessment process. However, the issue that resulted in the enforcement action will continue to be an input to assessment

- Assessment process results might be used to modulate enforcement actions (although assessment results would not affect the assessment of severity of the violation)

Further thought about the statements regarding RIBIP covering risk-significant attributes of licensee performance not adequately covered by PIs and verifying the accuracy of the PIs has lead to a modification in the original set of defining principles. Two additional principles represent this change:

- Adequate assurance of licensee performance requires assessment of both PIs and cornerstone inspection areas; PIs and cornerstone inspection areas deemed necessary for assessment can have equal importance in the assessment based on the risk significance of the performance issues

4. PI results may be overturned if inspections aimed at verifying their accuracy indicate significant weaknesses or when the licensee fails to report PIs

The principle related to thresholds has been expanded to reflect more detailed understanding of how thresholds will be used in the new process:

- Both the PIs and the cornerstone inspection areas will have established thresholds, risk-informed, where possible

5. Crossing of a PI threshold and a cornerstone inspection area threshold will have similar meaning with respect to safety significance; said another way, PI and cornerstone inspection area thresholds will be approximately equal in terms of their risk significance, where possible

6. An action-level will be set for unacceptable performance

Four bands of performance are envisioned for the data inputs, the:

- licensee response band, in which performance does not require NRC engagement, also called the “green” band in the NEI proposal;
- regulator response band, in which performance concerns prompt NRC engagement, in which the NRC would consider taking actions but in which action may default to the licensee with NRC oversight, also called the “white” band in the NEI proposal;
- required regulator response band, in which performance concerns prompt NRC action, also called the “yellow” band.
- unacceptable performance band, that requires an order to suspend, modify, or revoke licensed operations.

3 THE NEW PERFORMANCE ASSESSMENT PROCESS

The assessment process is continuous, in which data from PIs, inspections, and other sources (such as licensee self-assessments) are supplied for each plant and reviewed by NRC staff at the appropriate level. Periodically, this process is aggregated into more formal reviews which increase in level over a defined (annual) assessment cycle.

The assessment process has four main functions:

- collecting and integrating the data for each plant and comparing the data to thresholds to determine the level of performance achieved by a particular plant,

- taking action, which includes determining appropriate NRC actions based on a predefined logic model for the performance level achieved and obtaining NRC approval for the actions,

- communicating assessment and action determination results to NRC and licensee officials and to the public,

7. verifying action completion, which includes ensuring that required licensee actions have been completed and have corrected the performance concern.

Raw performance data is updated based on licensee submittal of PI data and branch chief (BC) submittal of plant issues matrix data (collection of inspection results). This data should be widely available to the licensee, public, and NRC staff throughout the assessment cycle. This availability will allow licensees and members of the public to view and comment on performance information on a continuous basis, and minimize the occurrence of surprises in the assessment process.

Further details regarding conducting the assessment, taking action, and verifying action completion are described in sections 3.2 through 3.4, respectively. The review processes for conducting these steps are described in section 3.1 below.

3.1. Review Mechanisms

NRC staff reviews of the raw and integrated data take place on different levels and frequencies: Continuous reviews are performed by inspectors, quarterly reviews by the Divisions of Reactor Projects' (DRP) BCs in the regions, mid-cycle and end-of-cycle reviews by regional management (division directors [DDs] and regional administrators [RAs]), and annual reviews to approve agency-level actions at the agency level. These are described in tabular form in Table 3.1 and in narrative form in the subsections below.

3.1.1. Continuous. A continuous review is conducted by inspection staff. This is an informal ongoing activity. Resident and region-based inspectors and various staff analysts will use the data to track performance in particular areas. Routine meetings held by the senior resident inspector (SRI) with the resident inspector (RI) staff can be used to perform this review. No formal assessment or communication to the licensee or public is expected.

Table 3.1. Review system.

Level of Review	Frequency/ Timing	Participants (* indicates lead)	Desired Outcome	Communication
Continuous	Continuous	SRI*, RI, regional inspectors, analysts	Performance awareness	None required
Quarterly	Once per quarter/ Two weeks after end of quarter	DRP: BC*, PE, SRI, RI	Input/verify PI/PIM data, detect early trends	Updated data set
Mid-Cycle	At mid-cycle/ Three weeks after end of second quarter	Divisions of Reactor Safety (DRS) or DRP DD*, DRP and DRS BCs	Detect trends, plan inspection for six months	Six month inspection look ahead letter
End-of-Cycle	At end-of-cycle/ Four weeks after end of assessment cycle	DRS or DRP DD*, RAs, NRR representative, BCs, principal inspectors, OE, OI, other HQ offices as appropriate	Assessment of plant performance, approve/ coordinate regional actions	Assessment letter and six month inspection look ahead letter
Agency Action Review	Annually/ Two weeks after end-of-cycle review	DIR NRR*, RAs, DRS/DRP DDs, AEOD, DISP, OE, OI, other HQ offices as appropriate	Approve/ coordinate agency actions	Commission briefing, followed by public meetings with individual licensees to discuss assessment results

3.1.2. Quarterly. A quarterly review is conducted by the DRP BC who has oversight responsibility for the facility, using branch resources. This is an informal data gathering and assessment activity, the primary purpose of which is to verify the accuracy of the quarterly data before releasing it to the public. It is triggered by the receipt of new data, which industry representatives have indicated could be made available to the NRC within approximately 15 days of the end of each quarter.

If significant changes in performance were identified, quarterly reviews could be used to trigger significant action. Typically, only small changes in the assessment inputs would be expected. Event follow-up is triggered separate from the assessment process and results are factored in as are any other inspection results. Communication to the licensee and public of quarterly assessment results is not expected, except as guided by the action matrix shown in Table 3.2. Public release of the quarterly update of the PI and inspection data is the outcome of this review.

3.1.3. Mid-Cycle and End-of-Cycle. Mid-cycle and end-of-cycle reviews are conducted by the region for each plant for which it has oversight responsibility. They consist of inspection planning reviews (for both) and a comprehensive assessment for the end-of-cycle review. It is expected that a rolling one-year window will be established for data inclusion, with information being provided regarding the age of the data (i.e., date of last update).

3.1.3.1. Mid-Cycle. The mid-cycle review is similar to the current regional resource planning meeting (plant performance review), and uses the data compiled during the previous 12 months. The mid-cycle review is used primarily to plan and assign inspection activities. Performance changes in the assessment inputs may be apparent, and may lead to the assignment of inspection activities beyond the RIBIP.

The mid-cycle review meeting will be chaired by a DRP or Divisions of Reactor Safety (DRS) DD or deputy DD, and will be staffed by members of the DRP and DRS branches responsible for directing inspection resources. RA involvement would not normally be required for the mid-cycle review, except as directed by the action matrix for significant changes in performance. The mid-cycle review will be held at approximately the six-month point in the annual assessment cycle (within three weeks of the end of the second quarter).

The deliverable from the mid-cycle review is a letter to the licensee, which details planned inspection activities for the next six months and indicates the reason for planned inspections outside the normal baseline inspections, if any.

3.1.3.2. End-of-Cycle. The end-of-cycle review is intended to be the comprehensive review of plant performance. It is similar in scope to the current SMM screening meetings. The purpose of the end-of-cycle review is to conduct a comprehensive assessment of licensee performance using all PI and inspection data and to plan inspection activities for the next six months. The BC will be responsible for providing the briefing package for this meeting. The briefing package for each plant will include; the threshold assessment and supporting data for each assessment input, a summary of the areas of concern, and an overall assessment of whether licensee performance is acceptable or unacceptable. It will also include a recommended action level, with justification as to how the recommendation was arrived at (i.e., an explanation that ties the recommendation to the decision logic model or action matrix).

The end-of-cycle review meeting will be chaired by a DRP or DRS DD or deputy DD. A senior manager from DRP, DRS, and NRR will attend along with the DRP BC and inspectors and BCs with oversight of significant inspections at the site throughout the cycle and the Office of Investigation and OE. The primary role of the HQ participants will be to provide perspective and ensure consistency in application of the process across regions and to ensure that assessment actions are consistent with other agency actions. Note that, because PI and cornerstone inspection area results are publicly available, all stakeholders have the opportunity to review the assessment data prior to the end-of-cycle review meeting, and can express concerns at any time during the process. This could be via written correspondence from the stakeholders to the NRC that documents their concerns. The results of the review will be presented to the RA for final approval. An inspection planning meeting with appropriate BC attendance may be conducted during or following the end-of-cycle review to determine the resources and schedules for inspections required during the next cycle.

Results of the assessment will be compared to the action matrix to determine appropriate actions to consider. Performance warranting agency-level action approval will be forwarded to the agency action review.

The deliverable from the end-of-cycle review is an assessment letter to the licensee. In addition to the planning information (six month plan), areas of NRC concern and any agency-wide actions will be included, as needed. Results will also be communicated via a public meeting, with the required level of NRC and licensee involvement being determined by the assessment results, as described in the action matrix. The assessment letters and public meetings will be released and occur, respectively, after the Commission meeting described below.

Plants needing agency-level action approval will be forwarded to the agency action review. Assessment letters and public meetings for plants not requiring approval of agency-level actions will be held until after completion of the agency action review meeting for all plants. A single Commission meeting will be held following the agency action review meeting to provide an opportunity for Commission review of assessment results and planned actions.

3.1.4. Agency Action Review. An annual review meeting is conducted by senior NRC managers, with the chairmanship by the Director of NRR, for plants needing additional review and approval of agency-level actions, as defined by the action matrix. The review uses data compiled by the previous reviews. Note that, because there are performance-based criteria that determine when a plant is subjected to the agency action review, it will not be held if no plants warrant agency-level actions; in that case, the assessment cycle is concluded with the end-of-cycle review. (The Commission briefing would still be held and assessment letters would still be issued after the briefing).

The agency action review involves a collegial review of plants requiring additional oversight due to adverse performance, with senior regional management presenting assessment results and proposed NRC actions for selected plants. The review will take place shortly (approximately two weeks) after the end-of-cycle review.

The purpose of the agency action review is to ensure a coordinated, balanced, and consistent agency response. Although the agency action review approval path is analogous to the SMM, it is different from the SMM in several critical respects:

- essentially the same materials used by the BC at the end-of-cycle review (with necessary modifications based on the outcome of that meeting) will be used in the agency action review; this should serve to greatly reduce the administrative burden of the current SMM, and

- the agency action review is reserved for plants where performance degradation meets certain criteria; there is no review of superior performers -- again, this should streamline the process and place emphasis on those areas requiring agency attention.

NRC actions will be based on the results of the assessment and will be consistent with guidance in the action matrix. Results of the assessment for plants discussed in the agency action review will be provided to licensees in an assessment letter. Annual results will also be communicated via a public meeting, with the required level of NRC and licensee involvement being determined by the assessment results. Communications for all plants will be released concurrently, following the agency action review.

3.1.5. Commission review. The EDO will give the Commission an annual briefing to convey the assessment results for *all* plants, with a focus on plants that required approval of agency-level actions, if any. The Commission will have negative consent on all assessment results and NRC actions prior to their release. The Commission review should occur within eight weeks of the end of the assessment cycle.

3.2. Conducting the Assessment

Conducting the assessment comprises two steps: organizing and compiling data and comparing grouped data to a threshold. Notice that each of these steps is applied on a per plant basis. That is, there is no provision for aggregating data across plants to compare one plant to another, provide for rankings, or even aggregate a list of all plants' assessments.

The organization and compilation of data implies that data serve as an input to the process. Because the team made some assumptions about the form and content of that data, as well as some decisions as to how to act in cases where data are not submitted as expected, receiving data is shown below as a process step.

In general, the process steps related to conducting the assessment (section 3.2) and taking action (section 3.3) would only occur in the context of the end-of-cycle review unless a significant change was noted in the status of an assessment input, as described in the discussions of quarterly and mid-cycle reviews above.

3.2.1. Receiving data. For the process to work, licensees must use a standard format, provided by the NRC, to report raw data and a threshold assessment (according to preestablished criteria) for each PI. These data will be reported quarterly, although it is acknowledged that some PIs may not change with that frequency. These data will be assumed to be valid unless RIBIP PI verification inspections indicate otherwise, as described in section 3.2.2 below. If the licensee fails to provide required data or inspection suggests that the PI data provided by the licensee is inadequate, the NRC will need to increase the scope of the RIBIP for that plant, thereby using inspection resources to collect data normally obtained through a PI. This response enables the NRC to obtain information necessary to assess performance in each of the cornerstones, even in those instances where PIs are inadequately reported.

Similarly, inspection results will be documented using a standard format, an adaptation of the current plant issues matrix, for forwarding inspection findings to the assessment process. As is the current situation, inspection data will be updated at the completion of the resident inspection report interval, or at the frequency it is collected (such as for special inspections). The formally issued PIMs should be updated at least as often as PIs (quarterly), although, as with the PIs, some data will be collected less frequently. Each item contained in the PIM will be identified with a code indicating to which cornerstone inspection area or PI it applies (depending on whether it is a primary data source or is being used to verify the accuracy of a PI), its significance, and context for the finding.

3.2.2. Organizing and compiling data. The new regulatory oversight framework provides the basic structure for organizing the data. Data will be grouped by cornerstone using the PI or cornerstone inspection area identifier. PIs are assigned to cornerstones by the framework team. An inspection area is also assigned to each cornerstone with individual inspection findings assigned to cornerstones by PIM codes, as discussed in section 3.2.1.

If RIBIP verification inspection findings verify the accuracy of a PI, assessment of the PI continues as usual. If inspection findings indicate a problem in collecting or reporting of the PI data, the RIBIP scope will be increased to include the PI area until such time as PI accuracy is restored. A similar approach would be taken in cases where a licensee failed to report data.

The BC is responsible for this process step, although delegation to the PE or SRI with BC oversight is possible. Data are reviewed as they are received, but at least quarterly, in conjunction with the quarterly review.

3.2.3. Comparing data to a standard. Changes in performance will be the determining factor in whether the regulator threshold has been crossed for each assessment input. PIs will have established thresholds (risk-informed, where possible) such that crossing into the white band will have one level of safety significance and crossing into the yellow band will have greater safety significance. The threshold values will be consistent with, although not directly mapped to, Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” where possible. See attachment 2 of the Commission paper for a detailed discussion of threshold development efforts.

Similarly, each individual inspection finding may have a significance categorization assigned to aid in determining if cornerstone inspection area thresholds have been crossed. Individual inspection findings will be aggregated by cornerstone inspection area. A group of low significance individual inspection findings (inspection observations, as defined below) alone would never be sufficient for crossing the yellow threshold in a cornerstone inspection area. The following represents a potential categorization scheme:

8. An **inspection observation** is defined as an NRC identified nonconformance with NRC requirements that has little or no safety significance. These items would be captured by licensee corrective action programs.

9. An **inspection finding** is defined as an NRC or licensee identified nonconformance that if uncorrected, would compromise the ability to meet the objective of the cornerstone linked to the inspection area. Cornerstone objectives could still be met pending completion of corrective actions with implementation of compensatory measures.

10. A **significant finding** is defined as an NRC or licensee identified nonconformance that if uncorrected significantly challenges the ability to meet the cornerstone objectives relative to public health and safety. Cornerstone objectives cannot be assured without completion of significant corrective actions.

This or an alternate categorization scheme will require further refinement. A process to use risk insights in assigning significance is under development.

Potential inspection area thresholds using this categorization scheme:

11. Green-white threshold for an inspection area: several (perhaps >3) inspection findings

12. White-yellow threshold for an inspection area: 1 significant finding or many (perhaps >10) inspection findings

(These thresholds are still under development, but must be consistent with the PI thresholds).

Using this scheme, the NRC can engage when any assessment input (PI or cornerstone inspection area) crosses the green-white threshold. Declining performance on multiple inputs to a single cornerstone or on the inputs to more than one cornerstone can then be used to trigger more comprehensive NRC action decisions, as described below in the action matrix.

3.3. Taking Action

In this portion of the assessment process the assessment results are translated into an associated action level. The “taking action” component comprises three steps:

- applying a decision model to produce an action level,

7. seeking approval for the recommended assessment and action level, and

H. communicating the assessment and action.

Note that the action matrix is not intended to provide guidance that is excessively rigid. It establishes expectations for interaction, licensee action, NRC inspection, and regulatory action decisions. It does not preclude less action or some additional action, when justified.

3.3.1. Apply a decision model (or an action matrix). The action level (from a range of possible actions) is determined by applying an action matrix as discussed below. This step is also performed by the branch chief, with oversight by the division director. Although it is expected that regions will use the action matrix to guide their actions in cases where crossing of a threshold is identified during quarterly and mid-cycle reviews of the data, formal application of the action matrix is required during the end-of-cycle review.

Table 3.2 shows the action matrix, which details the correspondence between licensee performance and NRC action. Recall that “white” is used as a shorthand way of saying that performance has crossed into the regulator response band and “yellow” means that regulator response is required. The matrix also shows the communications for given performance/ action levels.

Table 3.2. Action matrix.

		LICENSEE PERFORMANCE INCREASING SAFETY SIGNIFICANCE ---------->				
RESULTS		I. All Assessment Inputs (PIs and Cornerstone Inspection Areas) Green; Cornerstone Objectives Fully Met	II. One or Two Inputs White (in different cornerstones); Cornerstone Objectives Fully Met	III. One Degraded Cornerstone (2 Inputs White or 1 Input Yellow) or any 3 White Inputs; Cornerstone Objectives Met with Minimal Reduction in Safety Margin	IV. Repetitive Degraded Cornerstone, Multiple Degraded Cornerstones, or Multiple Yellow Inputs; Cornerstone Objectives Met with Significant Reduction in Safety Margin	V. Overall Red (Unacceptable) Performance; Plants Not Normally Permitted to Operate Within this Band, Unacceptable Margin to Safety
RESPONSE	Management Meeting	Routine Resident Inspector Interaction	SRI/BC Meet with Licensee	DD/RA Meet with Licensee Management	EDO Meet with Senior Licensee Management	Commission meeting with Senior Licensee Management
	Licensee Action	Licensee Corrective Action	Licensee Corrective Action with NRC Oversight	Licensee Self Assessment with NRC Oversight	Licensee Performance Improvement Plan with NRC Oversight	
	NRC Inspection	Risk-Informed Baseline Inspection Program	Inspection Follow-up	Inspection Focused on Cause of Degradation	Team Inspection Focused on Cause of Overall Degradation	
	Regulatory Actions	None	-Document Response to Degrading Area in Inspection Report	-Docket Response to Degrading Condition *(Consider N+1 Inspection for 2 Consecutive Cycles in This Range)*	-10 CFR 50.54(f) Letter - CAL/Order *(Consider N+1 Inspection for 2 Consecutive Cycles in This Range)*	Order to Modify, Suspend, or Revoke Licensed Activities
COMMUNICATION	Assessment Report	DD review/sign assessment report (w/ inspection plan)	DD review/sign assessment report (w/ inspection plan)	RA review/sign assessment report (w/ inspection plan)	RA review/sign assessment report (w/ inspection plan)	RA review/sign assessment report (w/ inspection plan)
	Public Assessment Meeting	SRI or Branch Chief Meet with Licensee	SRI or Branch Chief Meet with Licensee	RA Discuss Performance with Licensee	EDO Discuss Performance with Senior Licensee Management	Commission Meeting with Senior Licensee Management to Discuss Licensee Performance
				<---------- Regional Review I Agency Review ---------->		

Three characteristics of this matrix are worth noting:

(1) the list is organized to show actions from lowest to highest “severity” with the implication that there is a maximum action that would typically be taken at a given level, but that there is latitude for selecting a less severe action, depending upon the circumstances;

(2) there are overlaps in the actions possible for the various assessment outcomes, such that the most severe action for a more favorable outcome may be used as the least severe action for a less favorable outcome (for example, inspection follow-up may be selected as the appropriate action for multiple white PIs within a cornerstone);

(3) the revised assessment process highlights the possibility that performance may be unacceptable, which results in an order to modify, suspend, or revoke licensed activities as a maximum consequence.

As shown in Table 3.2, a combination of licensee action, NRC inspection, <u>and</u> regulatory action could be invoked when performance in any assessment input crosses the regulator response threshold. A more detailed decision logic will underlie the action matrix, to guide decisions regarding which of the possible range of actions is most appropriate for the situation under evaluation. Note that feedback regarding licensees’ failure to complete actions specified during the prior assessment period or licensees’ having taken actions that were unsuccessful in mitigating the concern is considered in making action determinations. Such cases may constitute cause for imposing more severe actions.

Principles to be embodied in the underlying logic include:

- Licensee action to address degrading performance should be considered first

- The effectiveness of the licensee’s corrective action and quality assurance programs will influence whether NRC action should be pursued

- NRC inspection beyond RIBIP will be performed in lieu of (or in addition to) licensee action if previous licensee actions have been ineffective in addressing degrading performance, or if the effectiveness of the licensee’s corrective action and quality assurance programs is in doubt

- Issuance of a 10 CFR 50.54(f) letter is usually the first regulatory action to take in response to systemic degradation of licensee performance

- If the licensee response to the 50.54(f) letter is adequate, then mutually agreed upon performance improvement metrics would be established

- Subsequent inspection planning meetings will determine actions necessary (e.g., NRC inspection, licensee self-assessment) to monitor the performance metrics so that licensee performance in addressing these commitments is reflected in future assessments

- Inadequate licensee response to a 50.54(f) letter will result in the issuance of a CAL, specific order, or order to modify, suspend, or revoke licensed activities. Mutually agreed upon performance metrics will be established for any licensee commitments made in the CAL or order. If mutually agreed upon performance metrics can not be developed, the staff can, if justified, impose them by order. Subsequent inspection planning meetings will determine actions necessary (e.g., NRC inspection, licensee self-assessment) to monitor the performance metrics so that licensee performance in addressing these commitments is reflected in future assessments

- If licensee actions taken to correct a white assessment input have been ineffective, as evidenced by the input remaining white during the subsequent assessment cycle, the NRC may consider more significant actions

When a plant is in an extended shutdown to address significant performance concerns, the plant will be removed from the normal performance assessment process. NRC Inspection Manual Chapter (IMC) 0350 will be used to monitor plant activities. Once the IMC 0350 process has been completed, the plant will be placed back in the normal assessment process.

Although existing regulatory vehicles were chosen for the regulatory actions, and many current actions were preserved, there are also noticeable differences from current practices. In particular, notice the absence of either a watch list or trending letter in either the action list or the needed communications discussed below. Direct recognition of "good performers" has also been eliminated, but balance is provided by communicating results for all plants, not just those that have problems. The action levels are graded based on licensee performance. As performance declines, the severity of actions increases.

Similarly, as shown in Table 3.2, communication with the licensee and with the public is graded. Grading takes place both with respect to the level of the communication and the level of NRC staff involved in the interaction. In all cases, it is assumed that the report of the assessment results (with a cover letter and supporting data) as well as an updated inspection plan will be sent to the licensee and the public document room. Regional involvement in all communications is assumed.

3.3.2. Approve the recommended assessment and action level. This step, which would be executed by the RA (but could be graded based on the assessment), serves as a final check on the appropriateness of the assessment and action level recommended by the BC. It is intended to ensure that:

- regional resources are sufficient to support the planned actions for all regional plants, and

- necessary interfaces have been conducted for actions that are outside the RA's independent sphere of influence.

The approval mechanism will vary, based upon licensee performance, as described in the above discussion of the end-of-cycle and agency action reviews. Plants having only one or two white assessment inputs will not be passed on to the agency action review. Plants having either two white or one yellow assessment inputs within a cornerstone, or any three white assessment inputs will not normally be passed on to the agency action review. For plants in this range (column III in the action matrix), Regional Administrators would only forward these plants to the agency action review if agency-level actions are proposed. These situations are expected to be rare. Plants in column IV will always be forwarded to the agency action review.

3.3.3. Communicate the assessment and action. Both the assessment and the resultant actions (with sufficient detail to understand how they were arrived at), will be communicated to the licensees, NRC management, the input providers (i.e., the inspectors), and the public in an assessment report.   Both the level of communication and the responsibility for these communications is defined by the action matrix. In general, the BC or DD communicates assessment results for cases in which no assessment inputs have crossed the regulator response threshold, the RA would communicate the results where the regulator response threshold has been crossed for multiple assessment inputs within a cornerstone, the EDO would communicate results related to the crossing of multiple assessment input thresholds in multiple cornerstones, and the Commission would be involved in communicating results that indicate a plant is unacceptable for operation. Additional communication will be taken as governed by the action matrix.

A standard assessment letter will be sent to all licensees annually. In some cases there may be a need for the licensee to respond in writing. A communication plan will provide additional details on a standard assessment report format, will specify the allowable time frame for responses, and will specify how responses will be handled. By way of example, however, the assessment report is expected to contain four critical sections as shown in Table 3.3.

Table 3.3. Assessment report critical sections.

1)	an overall statement regarding plant performance	Overall plant performance is acceptable.
2)	a statement of any areas of concern	The NRC notes degraded performance in the initiating events' area
3)	an enumeration of any "tripped" assessment inputs (PIs, cornerstone inspection area)	as evidenced by the regulator threshold having been crossed in PI XYZ (specific details would be given). No performance degradations were noted in other areas.
4)	a statement of actions to be taken	The NRC intends to conduct a regional initiative inspection focused on understanding the cause(s) of this degradation within the upcoming six month inspection cycle.

3.3.4. Consideration of licensee feedback. Both licensees and the public have an ongoing opportunity to provide comments on the data inputs to the assessment process because assessment inputs and their associated threshold determinations are publicly available, as noted in the discussion of end-of-cycle reviews. In addition, the licensee will be afforded the opportunity to comment on the actions to be taken. The assessment report will forward proposed actions to address degrading licensee performance. Actions taken in response to events or unacceptable performance (e.g., 10 CFR 50.54(f) letters or CALs) will be taken expeditiously, outside and separate from this process, as needed. In routine circumstances, the licensee will be given 30 days from the date of assessment report issuance to respond to the assessment results and proposed actions. Licensee input (such as a licensee proposal to perform self-assessment in lieu of NRC inspection) will be evaluated and final actions will be planned, communicated, and implemented.

3.4. Verifying Action Completion

As shown in the action matrix, most assessment outcomes will involve corrective action by the licensee. Whether corrective action has been taken, and whether it has been successful in correcting the safety issue that prompted the action, are important data points in determining NRC actions in subsequent assessments. Having the same assessment inputs persist in a state meriting regulatory engagement across assessment cycles warrants consideration of more severe actions, as shown in the action matrix.

The inspection program will provide the check on licensee actions. The inspection plan will be used to formalize verification assignments to RIs and SRIs, such that the continuous reviews provide status information on licensee completion of actions. Guidance regarding processes for verifying licensee actions will be developed. The BC will use that guidance to determine how and when completion of the actions by the licensee will be verified and will assign inspection resources accordingly.

4 PROCESS EVALUATION

Two types of evaluations of the new performance assessment process are planned. An initial benchmarking effort will be undertaken prior to and during the initial implementation of the new assessment process. Evaluations of the steady-state performance of the process will occur periodically once the new assessment process is fully implemented. Each of these evaluation types is discussed below.

4.1. Initial Benchmarking

Although full details of the initial benchmarking cannot be determined until the plan for transitioning from the current to the planned process has been fully developed and approved, it is expected that benchmarking of the assessment process will occur in several phases that track to a phase-in period for the new process, including:

- benchmarking of the individual PIs (to be conducted by the framework team),

- test application, planned for early, 1999, in which an initial trial of the workability of the proposed process, including ability to reliably assign risk significance and assessment area information to individual PIM entries, evaluate assessment input, cornerstone, and overall results, and reach conclusions related to actions to be taken that are consistent with actions suggested by concurrently or historically available independent data, will be conducted on a few plants, subject to the availability of the required assessment inputs,

- pilot (partial) implementation evaluation, planned for summer-fall, 1999, in which similar characteristics of the assessment process and results as detailed for the test application will be evaluated for a subset of plants taking place in a pilot or phased implementation, and

- full implementation evaluation, planned for summer, 2001, in which similar characteristics of the assessment process and results as detailed for the test application will be evaluated for all plants.

The full implementation evaluation will represent a segue to the steady-state evaluation process, and will likely involve considerations similar to those described in section 4.2, but is expected to be more comprehensive and formal than the steady-state evaluations.

4.2. Steady-State Evaluation

The steady-state evaluation comprises five main steps, each of which is discussed in detail below:

- aggregation of outcomes,
- evaluation of process compliance,
- evaluation of results against independent data sources,
- incorporation of process feedback, and
- objectives-based evaluation of process.

Feedback from the steady-state evaluation may affect almost any component in the overall process, depending on what process step is seen as contributing to identified problems. Feedback from this sub-process could also affect the inspection program, if inadequacies in that program were identified as causing assessment-related problems, as well as the enforcement program, if plants' enforcement histories were found to be systematically different from their assessment results.

It is not intended that the steady-state evaluation would occur after every assessment cycle (except, perhaps, in the first year following complete implementation). Rather, it would occur with somewhat lower frequency -- every two years is the recommended cycle. Or, it could be triggered by the presence of new information or priorities that might have an effect on the conduct of assessments.

Detailed development of how the steady-state evaluation would be implemented has yet to be undertaken. Initial thinking regarding what would be accomplished at each step of the process is documented below.

4.2.1. Aggregation of outcomes. This step mainly implies a compilation of the results of the assessments conducted during the review cycle. The aggregation should result in the organization of data across several assessment cycles such that patterns that may not have been picked up in a single assessment cycle can emerge.

4.2.2. Evaluation of process compliance. This step is intended to communicate the need for examination of the documented assessments to determine whether there have been deviations from the process guidance in sufficient numbers to consider that there may be some systemic process flaw at the root of the deviations. Of particular interest are repeated overrides of the action matrix by the decision authority, which may indicate that the action levels are inappropriate. Repeated requests for additional data needed to complete the assessment (such as additional inspections or licensee self-assessments), which may be indicative of problems in the data collection strategy, will also need to be evaluated.

4.2.3. Evaluation against independent data sources. Results of the assessment process will be compared to the results of other evaluations, including sources such as previous NRC assessments; accident sequence precursor analyses; enforcement histories; assessments conducted by other organizations; protests of results by licensees; and stakeholder assessments, such as that represented by public interest groups. One concern in performing comparisons with externally provided data is the quality and currency of that data, so one might add the caveat that external comparisons would be performed as long as data quality could be verified and the same data had not been used in a previous steady-state evaluation.

In cases where discrepancies are identified, an attempt will be made to understand the sources of the discrepancy. Causes could include: differences of opinion due to evaluation of different data; NRC assessments being of higher severity due to consequences of the licensee's failure to take action from previous assessments or to take unsuccessful actions; or NRC error. The latter is expected to be rare, and will not result in a retrospective adjustment to assessment results. It will, however, result in a requirement to examine the adequacy of the assessment inputs, the decision process model, and/or the implementing guidance to understand and correct factors that may have led to the error.

4.2.4. Incorporation of process feedback. There are many stages throughout the overall assessment process at which stakeholder input is possible. For example, licensees may question the significance of inspection findings or assessment results; inspectors or licensees may indicate difficulties in providing (completing or reporting) required assessment inputs; or public comment regarding the scrutability or appropriateness of assessment results could be received as a result of public meetings to communicate findings. In this step, that input is systematically evaluated for trends -- especially cases where diverse stakeholders identify the same process problems -- and consideration is given to whether process changes are warranted. In addition, there may be new information (such as new NRC priorities) that might be considered with respect to its implications for process changes.

4.2.5. Objectives-based evaluation of process. The process and all of the evaluations performed on it in the previous steps are compared to the process objectives and criteria to determine whether the process:

- meets its stated objectives in its “as is” condition,

- could be expected to continue to meet (or resume meeting) its stated objectives were it modified consistent with recommendations obtained through either the evaluation of process compliance or the incorporation of process feedback (or both).

Table 4.1 below shows the objectives and criteria that were developed for the integrated assessment process, with changes to be consistent with the new performance assessment process. The table is provided as an example of the types of objectives and criteria that can be considered.

Table 4.1. Success criteria for the assessment process.

Objective Being Measured	Success Criteria and Measurement Methods
Simplicity/non-redundancy/efficiency	• Task analysis on new process shows not more than *10%* non-compliance with activities as assigned per process diagram (by position or by the addition of tasks) • Pre-post survey of NRC staff satisfaction shows statistically significant improvement
Resource constraints (which also speak to efficiency)	Level of effort spent on assessment is *25%* less than with current process, as indicated by the regulatory inspection tracking system for RIs and SRIs and estimates of time spent for BCs and above
Scrutability and data integrity	Number of executive over-rides (cases where the outcome is something different than the input) at end-of-cycle review is less than *5%*
Consistency (repeatability)	Inter-rater reliability of *95%* on categorization and assignment of risk significance of the PIM and execution of the process model (threshold assessment and action determinations)
Validity	Plant peer group predictive validity correlation on assessment results and actions taken (actions should be consistent with performance)
Scrutability	Pre-post survey of stakeholder perceptions of process scrutability shows statistically significant improvement
Timeliness	• *95%* of available data is documented within *two weeks* of the end of the assessment period • *95%* of end-of-cycle reviews occur within *four weeks* of assessment period end • *95%* of agency action reviews occur within *two weeks* of end-of-cycle reviews • *95%* of Commission meetings occur within *eight weeks* of the end of the assessment cycle • *95%* of letters to licensees out within *one week* of Commission notification and within 60 days of the end of the assessment period
Data integrity	Number of instances in which retrospective review of assessment results compared to independent data sources reveals discrepancies is less than 5%
Predictiveness	Over time, the number of severity level III and higher violations, the number of yellow assessment inputs, and the number of plants referred to the agency action review decrease

ENFORCEMENT PROGRAM CHANGES

Attachment 5

1 INTRODUCTION

The purpose of the NRC's enforcement program is to support the NRC's overall safety mission in protecting the public and the environment. Consistent with that purpose, enforcement action is used as a deterrent to emphasize the importance of compliance with requirements and to encourage prompt identification and prompt comprehensive correction of violations. Thus, enforcement is a regulatory tool that supports the NRC inspection and assessment process by reinforcing the need to avoid violating requirements and emphasizing the need to identify safety issues and to implement lasting corrective actions. It does not appear that the purposes of the NRC power reactor enforcement program need to be changed as a result of changes in the inspection and assessment processes. However, changes will need to be made in severity levels to align the process for evaluating the significance of violations with the process for evaluating the significance of inspection findings and performance indicator results. In addition, the civil penalty assessment process may need to be changed. Preliminary views are discussed below on how the Enforcement Policy and program for power reactors might be changed. However, it is premature to develop specific changes until the assessment process is more developed.

The first step in the enforcement process is to understand the significance of the violation or grouping of violations to determine an appropriate enforcement sanction. The process used to understand the significance of an issue uses many of the same attributes as the assessment process. In the same manner as the proposed assessment process, violations are assessed for significance and a licensee's past performance is considered in determining the adequacy of the licensees' actions in addressing and correcting past violations.

2 PROPOSED APPROACH

It is important that there be close alignment between the enforcement and assessment processes because of the similarities between them. A significant issue in the assessment process should be significant in the enforcement policy. It is premature to develop specific changes to the enforcement process at this time due to the ongoing efforts to make improvements to the inspection and performance assessment processes for power reactors. However, the Office of Enforcement staff is considering the following:

11. Enforcement should not drive the assessment process. The thresholds for significance of issues should be the same in both assessment and enforcement programs. The assessment process thresholds should set the standard and the enforcement process should ensure that the sanction given is in keeping with the significance from an assessment perspective. To accomplish this, a number of examples in the supplements of the Enforcement Policy will need to be modified to be aligned with the assessment process.

12. The proposed third exception in SECY 98-256, "Proposed Revision to the Enforcement Policy to Address Severity Level IV Violations at Power Reactors," for issuing a notice of violation for a violation that it is repetitive as a result of inadequate corrective actions and identified by the NRC, would be deleted. Repetitive violations would be treated as part of the assessment process, in that the NRC would not place additional focus on

such violations by requiring a response unless performance crossed into the regulator response band.

13. A decision will need to be made on the treatment of violations that are currently aggregated, based on safety significance or regulatory significance, to a severity level III. Several options are being considered. Other options may be developed.

    Option 1 - Establish the severity of violations based on the as found condition. If the violation or grouping of violations associated with a single issue or event in themselves based on actual or potential consequences are not significant, the violations should be treated as individual seventy level IV violations regardless of their regulatory significance. NOVs or NCVs would be issued based on the policy for Severity Level IV violations. The assessment process would be used to reach the root causes or underlying issues which are of regulatory significance.

    Option 2 - As in Option 1, establish the severity of violations based on the as found condition. If the violation or grouping of violations associated with a single issue or event in themselves based on actual or potential consequences are not significant, the violations should be treated as individual seventy level IV violations regardless of their regulatory significance. NOVs or NCVs would be issued based on the policy for Severity Level IV violations. In addition, however, if the licensee is performing in the regulator response band, a written response by way of a NOV could be required for such violations to leverage NRC resources. This process would replace the third exception in SECY 98-256 (see item 2 above). A licensee performing in the licensee response band would not have to respond in writing to Severity Level IV violations unless the other criteria of SECY 98-256 is met.

    Option 3 - This option would preserve the concept of regulatory significance. Two sub-options might be considered. Option 3A - Individual Severity Level IV violations could be aggregated into a Severity Level III problem if the violations are significant enough in the assessment process to cause a licensee to cross from the licensee response band to the regulator response band. Option 3B - Alternatively, those individual Severity Level IV violations could be treated as another exception to the non-cited violation (NCV) policy for Severity Level IV violations resulting in a written response.

    Resolution of this issue needs to await a better understanding of the assessment process and how that process will be treating inspection findings.

14. The above approach for aggregating violations to a higher severity level would not be applicable to violations associated with an individual's integrity or with impeding the regulatory process, (e.g., willfulness, 10 CFR 50.9, 50.59, 50.72 and 73, and 50.7). These violations could be considered escalated based on the regulatory significance, regardless of the licensee's performance band.

15. The civil penalty assessment process could be simplified by removing the past performance decision criterion related to performance in the past two years, and focusing only on whether the licensee identified the issue and took appropriate

corrective action. The reactor assessment process might be the more appropriate place to consider past performance.

16. In addition, improved guidance is needed for the criteria for categorizing violations as minor. Since a minor violation is not normally documented in an inspection report, the needs of the reactor assessment process should set the minor violation criteria.

TRANSITION PLAN

Attachment 6

1 INTRODUCTION

This attachment provides the plan to be used by the NRC to transition through the implementation of the revised oversight process. The transition plan includes change management strategies for creation of management systems necessary to support those desired changes. Together, these aspects are key ingredients in enabling an organization to successfully implement change.

The transition plan contains milestones for both the NRC and industry. Successful implementation will require a continuing interface with the industry and other stakeholders at various stages. Significant investment in staff and management resources will be required to complete necessary supporting documents and infrastructure, develop and train staff, and manage all aspects of the resulting change effort.

The transition plan contains challenging but achievable goals. The milestones reflect best estimates based on recognized challenges. Adjustments will be made as necessary to allow for resolution of unanticipated problems (e.g., difficulty in assigning significance to inspection findings, difficulty in collecting PI data in a consistent manner, unexpected change in resources) or additional direction from the Commission.

2 DISCUSSION

The attached table shows the major milestones for the transition, creating a shared vision, and creating management systems to support the change. “Plan/process development” identifies the objectives and the “Implementation” identifies those necessary activities to support those objectives. “Training” and “Communication” identifies those activities necessary to foster and reinforce a shared vision. “Phase-in with Existing Processes” identifies those interfaces with current processes.

Successful implementation is predicated on several key assumptions, including that:

- The Commission approve the revised oversight process without major changes by March 31, 1999

- The pilot process (baseline inspection program, PI collection) begins by June 1999

- The successful benchmarking of selected plants to new process by March 1999

- The successful establishment of the “Change Coalition” and “Transition Task Force” by February 1999

2.1. Pilot Program

The pilot program will be conducted using two plants from each region including both PWRs and BWRs . The NRC will coordinate with the industry to identify those plants that will use the risk-informed baseline inspection program in lieu of the current process. Draft inspection procedures and approved performance indicators will be developed prior to beginning the baseline inspection program for the pilot plants. The 6-month pilot program will be measured against previously established criteria, and full implementation will begin if the success criteria are met. The staff used this structure to allow for transition to full implementation, upon successful completion of the pilot program.

2.2 Communication

Communication of the new oversight process for both internal and external stakeholders will begin with a high level vision of the approach and overall direction. Subsequent training will result in a progressively more detailed fashion. A highlight of the communication effort is an in-depth training of all inspectors and affected managers at a joint NRC and industry workshop. Periodic press releases will be released as major milestones are achieved during implementation of the process.

2.3. Phase-in with Existing Processes

SALP suspension will continue indefinitely and the SALP program will eventually be discontinued upon successful implementation of the pilot program. Plant Performance Reviews (PPRs) will continue on a six-month cycle until replaced by Mid-Cycle and End-Of-Cycle Reviews. Senior Management Meetings (SMMs) will continue on a yearly spring cycle until replaced by Agency Action Reviews.

3 CHANGE COALITION

A key factor during the implementation of the new process focuses on creating and maintaining a shared vision within the NRC. “Opinion Leaders” are individuals within the organization who have significant credibility among their peers such that their peers’ views are influenced by the opinion leader’s views. The identification and cultivation of opinion leaders at both the regional and Headquarters offices will be important for creating alignment within the agency and extending that vision to other stakeholders. These opinion leaders will be the “agents of change” within the NRC and will form the “Change Coalition.” The Change Coalition will be the communication ambassadors at all levels within the agency. This group will highlight why changes are necessary, what the changes will be, and how change will be effected. It is anticipated that the industry will be conducting a similar process during program implementation. Future meetings between the NRC and industry will focus on the progress of the culture change and any recommendations to improve the process.

4 TRANSITION TASK FORCE

A Transition Task Force (TTF), which is separate from the Change Coalition, will be formed in order to manage the phase-out of the existing process and the phase-in of the new oversight process. The role of the TTF will be to complete the development of detailed implementing instruments and infrastructure. Although the staff anticipates overall resource savings once these recommended program changes are fully implemented, there will be considerable resources required to implement these changes. The staff estimates approximately 17-19 FTEs will be required to develop and implement the recommended changes. This is in addition to the 6.5 FTEs expended to date in FY 1999 for the development of these recommendations. These FTEs are within the currently budgeted resources in FY 1999 and FY 2000 for developing and implementing changes to the inspection and assessment programs. These activities have been accounted for in the Reactor Performance Assessment Program and Inspection Program Operating Plans.

4.1. Process Development/ Implementation

- Develop Pilot Plan, including success criteria
- Review and incorporate public comments, revise process as appropriate
- Develop individual baseline inspection program inspection procedures (IPs) for pilot use (8-10 IPs)
- Finalize guidelines for establishing significance of inspection findings
- Develop modifications to Inspection Manual Chapter (IMC) 2515 *Light-Water Reactor Inspection Program- Operations Phase* - revise program guidance
- Develop modifications to IMC 0610 *Inspection Reports* - report writing
- Quantify change in burden based on implementation of the new process (SRM M981102)
- Develop replacement to the PPR IMC, including PIM guidance and inspection planning
- Develop a new assessment process Management Directive
- Identify computer infrastructure needs
- Establish data (PI) format needs and definitions
- Benchmark process against prior plants
- Oversee pilot implementation
- Develop necessary changes to draft documents listed above

4.2. Training/Communication

- Conduct training for all inspectors
- Conduct a pilot utility workshop
- Develop required training
- Conduct regional meeting with stakeholders
- Conduct a joint NRC/ industry pre-implementation workshop

DRAFT

Revised Regulatory Oversight Process
- Transition Plan -
1/4/99

Date	Plan/Process Development	Training	Implementation (Revised Process)	Communication	Phase-in with Existing Processes
Dec 98	12/18 - Complete Scoping of Transition Task Force Activities (Detailed Process Development, Pilot Implementation; Phase-in; Interface Issues) 12/24-Comm. Paper to EDO		12/18 - Identify Change Champion and Change Coalition (CCL)	12/1 - Brief Regional DRS Directors 12/3 - Brief ACRS 12/8 - Brief Oversight Steering Committee 12/16 - NRC/NEI Meeting to Discuss Final Process Details	12/17 (Week of) - Issue Guidance for Format/Content of PPR Letters (Modified to Support SALP Suspension)

DRAFT

Revised Regulatory Oversight Process
- Transition Plan -
1/4/99

Date	Plan/Process Development	Training	Implementation (Revised Process)	Communication	Phase-in with Existing Processes
Jan 99	1/8 - Commission Paper to OCM 1/11- Begin Benchmarking of Selected Plants (Process) 1/22 - Pilot Plants named 1/22 - FRN Issued to begin Public Comment Period 1/29 - Define Inspection Program Organization (Approach to Grouping Similar Tasks Into Procedures, Identifying Inspector Job Responsibilities, etc.) 1/26 - Draft Pilot Plan Completed		1/15 - Transition Task Force* (TTF) Named 1/22 - 30 day public comment period begins	1/14 - Brief Regional DRP Directors 1/14 - Meet with NEI to Discuss Pilot Plan 1/20- Commission Briefing on Process Recommendations 1/22 - Press Release to Announce 30 Day Comment Period 1/25 (week of) - Meeting Conducted with CCL (roles, expectations, buy-in) 1/26 Brief ACRS on Final Recommendations 1/28 - Brief Industry Regulatory Compliance and Technology Group	

DRAFT

Revised Regulatory Oversight Process
- Transition Plan -
1/4/99

Date	Plan/Process Development	Training	Implementation (Revised Process)	Communication	Phase-in with Existing Processes
Feb 99	2/16 - Transition Task Force Activities Begin (e.g., IP Development, etc.) 2/26 - Complete Process Benchmarking for Selected Plants 2/26 - Develop Commission Paper to Provide Results of Public Comment		2/16 - TTF in place	2/1 (week of) - Meeting of CCL 2/2 - NEI Meeting with Industry; Site VPs/Licensing Managers - East 2/3 - NEI Meeting with Industry; Site VPs/Licensing Managers - West 2/11 NEI Task Force Briefing of NSIAC 2/23 - Public Comment Period Ends Regional Meetings (coincident with PPRs to describe new process)	PPRs Conducted Using Existing PIMs

DRAFT

Revised Regulatory Oversight Process
- Transition Plan -

1/4/99

Date	Plan/Process Development	Training	Implementation (Revised Process)	Communication	Phase-in with Existing Processes
Mar 99	3/19 - Commission Paper Forwarded to the Commission (Results of Public Comments) 3/26 - Development of Draft Procedures Completed 3/31Receive SRM on Proposed Staff Changes.			3/3-5 Regulatory Information Conference (Introduce High Level Concepts) 3/26 - Draft IP and IMC 0610 & PIM Guidance for Pilot Use Issued for Comment (made available to the public)	SMM Screening Meetings
Apr 99	4/9 - PI Data Format Established (draft) (NRC/NEI)			4/6-8 - Briefing for American Power Conference	4/21-22 - SMM

DRAFT

Revised Regulatory Oversight Process

- Transition Plan -

1/4/99

Date	Plan/Process Development	Training	Implementation (Revised Process)	Communication	Phase-in with Existing Processes
May 99	5/14 - PI Reporting Format Finalized (NRC/NEI) 5/17-Issue Letter to Pilot Plants to Initiate Pilot Process 5/21 - Process and Procedure Guidance Finalized for Pilot Use	5/10 - Joint Utility/NRC/NEI meeting for Pilot Plant Training (Mgr/BC/PE/SRI)		5/24 - Joint NRC/NEI Meeting to Resolve Issues Prior to Pilot	Brief Commission on SMM results
Jun 99	6/1 - Issue Revised Enforcement Guidance	6/15 - Provide Training to the Regions on Enforcement	6/1 - Begin Pilot Process (baseline inspection program, PI collection)(previous two years, monthly data)	6/15 - Issue Press Release on Enforcement Revisions	
Jul 99	7/30 - Check-in with Pilot Utilities and Regional Implementors (NRC/NEI)		7 /15 - First PI Data Submitted (Monthly for Pilot Only) IRs Issued	7/15-30 Conduct Regional Meetings with States on details of new process	

DRAFT

Revised Regulatory Oversight Process

- Transition Plan -

1/4/99

Date	Plan/Process Development	Training	Implementation (Revised Process)	Communication	Phase-in with Existing Processes
Aug 99			8/15 - PI Data Submitted IRs Issued 8/27 - Pilot Plant Quarterly Review Conducted		PPRs Conducted Using Existing Inspection Program/Process
Sep 99	9/10 - Review of Pilot (NRC/NEI) - consistency - data quality - insp. results - enforcement 9/24 - Develop Revisions to the IPs and Process Based on Pilot Lessons 9/24 - Develop any Proposed Organization Re-alignment to Support Implementation of the Revised Process		9/15 - PI Data Submitted IRs Issued	Brief Commission TAs on Progress	

DRAFT

Revised Regulatory Oversight Process

- Transition Plan -

1/4/99

Date	Plan/Process Development	Training	Implementation (Revised Process)	Communication	Phase-in with Existing Processes
Oct 99	10/4 - IPs and Reporting Format and guidance Finalized	10/11-25 - Conduct training of all Reactor Program Staff on New Process (coincident with PPRs) 10/11-25 - Conduct Training for Regional Branch Chiefs on Process Implementation	10/1- PI Data Submitted IRs Issued	10/11-25 (TBD) - Conduct Joint NRC/Industry 2-day Workshop (NRC/NEI) Issue a Press Release Regarding the Workshop	
Nov 99			11/15 - PI Data Submitted IRs issued 11/30 Pilot Plant End-of-Cycle and Agency Action Review Conducted (Using 12 month PI Data, 6 Month Inspection Results)		

DRAFT

Revised Regulatory Oversight Process
- Transition Plan -

1/4/99

Date	Plan/Process Development	Training	Implementation (Revised Process)	Communication	Phase-in with Existing Processes
Dec 99	12/15 - Final Review of Pilot (NRC/NEI) - consistency - data quality - insp. results - enforcement 12/15 - Measure Process Performance for Pilot Plants Against Success Criteria. If met, Proceed with Full Implementation.			Brief Commission TAs	
Jan 00	1/1 - Revised Oversight Process Becomes Effective for All Plants 1/1 - MD on SALP Deleted		1/1 - Risk Informed Baseline Inspection Program Implemented for all Plants 1/1 - PI Data Reporting Begins for All Plants	1/15 - Press Release Issued Announcing Full Process Implementation and SALP Deletion	1/1 SALP Discontinued
Feb 00					PPR Conducted
Mar 00					Conduct SMM Screening Meetings
Apr 00			Conduct Quarterly Review		Conduct Last SMM
May 00					

DRAFT

Revised Regulatory Oversight Process

- Transition Plan -

1/4/99

Date	Plan/Process Development	Training	Implementation (Revised Process)	Communication	Phase-in with Existing Processes
Jun 00	6/15 - Issue MDs on new process 6/30 - Review Process Implementation (NRC/NEI)				
Jul 00			Conduct Quarterly Review		
Aug 00					
Sep 00					
Oct 00			Conduct Mid-Cycle Review		
Nov 00					
Dec 00					
Jan 01			Conduct Quarterly Review		
Feb 01					

DRAFT

Revised Regulatory Oversight Process
- Transition Plan -

1/4/99

Date	Plan/Process Development	Training	Implementation (Revised Process)	Communication	Phase-in with Existing Processes
Mar 01					
Apr 01			Conduct End-of-Cycle Review Conduct Agency Action Review		
May 01	5/15 - Review Process Implementation			Commission Briefing on Assessment Results Press Release Issued	
Jun 01	6/25 - Conduct Process Review. Identify Lessons Learned. Develop Appropriate Modifications.		6/7-End of Cycle Letters sent 6/30 - Implement Lessons Learned From Assessment		

*TRANSITION TASK FORCE - TASKS

Process Development/Implementation

- ☐ Develop Pilot Plan, including success criteria (Moderate)
- ☐ Develop individual baseline inspection program inspection procedures for pilot use. (8 - 10 IPs) (Substantial)
- ☐ Finalize guidelines for establishing significance of inspection findings (Moderate)
- ☐ Develop modifications to IMC 2515 - revised program guidance (Moderate)
- ☐ Develop modifications to IMC 0610 - report writing (Substantial)
- ☐ Review and incorporate public comments, revise processes as appropriate (Small-Moderate)
- ☐ Quantify change in burden based on implementation of the new process (SRM M981102) (Moderate)
- ☐ Develop replacement to the PPR IMC, including PIM guidance and inspection planning (Moderate)
- ☐ Develop a new assessment process Management Directive (Moderate)
- ☐ Identify computer infrastructure needs (Moderate)
- ☐ Establish data (PI) format needs and definitions (Moderate)
- ☐ Benchmark process against prior plants (Substantial)
- ☐ Oversee pilot implementation (Moderate)
- ☐ Develop necessary changes to draft documents above (Moderate)

Training/Communication

- ☐ Develop required training (Substantial)
- ☐ Conduct training for all inspectors (Substantial)
- ☐ Conduct pilot utility workshop (Small)

- ☐ Conduct joint NRC/industry pre-implementation workshop (Substantial)

- ☐ Conduct regional meeting with stakeholders (Moderate)

SUMMARY OF IRAP PUBLIC COMMENTS

Attachment 7

1/4/99

Background

On March 9, 1998, the staff issued SECY-98-045, “Status of the Integrated Assessment Process for Operating Commercial Nuclear Reactors,” which forwarded the staff’s recommendation for a new integrated assessment process and requested the solicitation of public comment on the proposed process. On June 30, 1998, the Commission approved the solicitation of public comment. This attachment summarizes the public comments.

General Information

The 26 respondents to the *Federal Register* notice are listed in attachment 7a. Industry groups submitted 19 of the 26 responses. Industry groups are represented by the Nuclear Energy Institute (NEI), licensees, support contractors, and law firms representing the interests of licensees. The licensee respondents operate 60 U.S. commercial nuclear power plants (slightly more than half of all U.S. plants). Public advocacy groups submitted 3 of the 26 responses , 3 of the 26 responses came from concerned citizens or consultants, and one of the 26 responses came from a State government.

The scope of the responses varied considerably among the respondents. Some addressed every subject in the questionnaire, others wrote general letters that did not specifically address the questions in the *Federal Register* notice. Because the responses were so varied, the staff chose to capture general comments from the responses (primarily extracted from the text of the letters) as well as specific responses to the questions. Comments that differed from the majority opinions are listed under “Other views” . **Specific responses are numbered according to their originator as listed in Attachment 7a.**

These public comments were evaluated and considered during the development of the regulatory oversight improvements. Public input is appropriately reflected in the recommended changes to the inspection, assessment, and enforcement processes. As stated in the *Federal Register* notice, individual comments may be reviewed at the NRC Public Document Room.

A. **REGULATORY OVERSIGHT APPROACH**

1. The NRC currently has a low threshold for initiating increased interaction with licensees above the core inspection program. For example, procedure adherence errors or program implementation weaknesses with low actual safety consequence may result in increased inspection activity in these areas. Alternatively, if these regulatory oversight thresholds were raised, the NRC would wait until actual safety significant events occurred (such as those measured by performance indicators) before increasing interaction with licensees.

    a. **At what threshold should the NRC take action to assure the adequate protection of public health and safety?**

    General comments

All the 19 industry representatives responded that the threshold should be based on objective, measurable performance indicators that relate to protecting public health and safety as detailed in the NEI proposal. One respondent stated that the NRC must establish a threshold upon which an underperforming plant must be shut down. Another respondent stated that the NRC should set the threshold for taking action at the compliance level.

Other views

The threshold should be at the compliance level. The NRC's response to a compliance violation should be appropriate to the importance of the violation to public health and safety. The NRC should take action if regulations are violated. If the NRC considers particular violations to be unimportant to pursue, the NRC should consider revising, by rule, the regulations that were violated (5).

The NRC should take action prior to a degradation in the licensing basis that results in a diminution of the defense in depth provided at each reactor (13).

b. **What is the basis for this threshold?**

General comments

The 19 industry groups stated that the basis for the threshold levels depends on the specific performance indicator. The threshold should be a blend of regulatory requirements (as set out in technical specifications and regulatory guidance) and risk insights.

Other views

The appropriate threshold for triggering an immediate plant shutdown should be the discovery of a degraded condition which does, or could challenge the public protection guidelines of 10 CFR 100 or plant worker protection guidelines of 10 CFR 20. The simple fact that these guidelines have not been exceeded is the wrong basis for continued operation of any nuclear facility (3).

2. **What range and specific types of NRC actions should be taken if licensees exceed the regulatory thresholds discussed in Question A.1?**

General comments

The industry groups stated that NRC responsive actions should be commensurate with the degree to which the violation endangers public health and safety. NRC response may range from relatively mild actions for events of low risk significance to the issuance of orders and civil penalties for events of high risk significance. Two respondents recommended that shutdown orders must be within the NRC's range of actions. One respondent added that when several licensees exceed the same threshold the NRC must initiate broader actions.

Other views

The NRC has a range of appropriate regulatory tools at its disposal. However, recent attempts to do away with level four violations is a step in the wrong direction (13).

3. The current regulatory oversight process focuses discretionary inspection resources on a selective sample of all aspects of licensee performance, such as human performance, procedure quality, and program implementation.

   a. **Could an enhanced use of high level performance indicators (e.g. operational transients and safety system availability) reduce the need for discretionary inspection if particular levels of licensee performance are achieved?**

General comments

The majority of respondents supported the use of performance indicators (PIs). Two respondents would not support the use of PIs if their use eliminated or mitigated an NRC response once a threshold had been violated. Two respondents did not believe that the use of PI's should replace in-depth inspection activities.

   b. **Would this approach result in a regulatory oversight process which is timely and comprehensive enough to ensure the adequate protection of the public health and safety?**

General comments

The majority of respondents stated that the use of PIs would be timely and comprehensive enough to ensure the protection of public health and safety. One commenter dissented.

Other views

High level performance indicators are not a substitute for in-depth inspection and evaluation of the plant against regulations. The NRC's use of SALP, PIs, and numerous studies of inspection reports have not, in many cases, led to conservative actions that could have been taken before *safety issues were brought to light by outside forces such as whistle blowers* (5).

4. **What should the role of licensee audits, inspections, and self-assessments be in the regulatory oversight process?**

General comments

The majority of respondents supported the enhanced use of licensee self-assessments (and corresponding reduction in NRC discretionary inspection). This would allow the NRC to focus its inspection resources on areas not covered by the licensee and on areas in which performance indicators show poor performance. Two respondents supported this position as long as the rigor and independence of licensee audits, inspections, and self-assessments could be validated and that the reports be made publicly available. There were two dissenting opinions.

Other views

Licensee audits, inspections, and self-assessments should be used as information sources in the assessment process. NRC should validate licensee information only to the extent required to develop confidence in its accuracy (9).

They should not be used in the oversight process to limit NRC inspection or mitigate NRC action. If industry wishes to rely upon assessments by owners groups, or INPO, these documents must be made publicly available (13).

5. **Would an enhanced use by the NRC of licensee audits, inspections, and self-assessments (and a corresponding reduction in NRC discretionary inspection) result in a regulatory oversight process that was sufficiently independent?**

General comments

The majority of respondents stated that the use of licensee audits, inspections, and self-assessments would result in a regulatory process that was sufficiently independent. There was one dissenting opinion.

B. **INTEGRATED ASSESSMENT PROCESS**

1. Objectives and Attributes

a. The objectives developed by the staff for an integrated assessment process include the following: (1) provide early warning of declining licensee performance and promote prompt, and timely corrective action; (2) provide checks and balances with other processes; (3) allow for the integration of inspection findings and other relevant information; (4) focus NRC's attention on those plants with declining or poor performance; (5) effectively communicate assessment results to the licensees and the public; and (6) allow for effective resource allocation. **What changes could be made to these objectives and why?**

General comments

The industry supported the basic process objectives but stated that the purpose and objectives were more completely stated in the NEI proposal . The industry stated that (1) assessment should drive the inspection focus and resource allocation and (2) enforcement should be driven by non-compliance with requirements that have safety significance and should not be an input into the assessment process. One respondent stated that the following two additional process objectives must be added: (1) provide objective criteria for prompting the shutdown of a nuclear power plant with inadequate safety margins and (2) eliminate subjectivity from the assessment process to the maximum extent possible. Two respondents fully supported the proposed objectives and attributes.

b. The new integrated assessment process would not formally recognize superior licensee performance, nor would it include a "watch list". **Should the NRC recognize superior licensee performance?**

General comments

All respondents agreed that the NRC should not formally recognize plants that perform at a superior level. The industry did not support the continuation of the watch list. Two respondents supported the continued use of the watch list.

Other views

The NRC should not discard its watch list until the revamped assessment process has been fully implemented and demonstrated successful for at least two complete assessment periods. However flawed, the watch list forms part of the bridge to the future assessment process (3).

A vehicle like the watch list sends an important signal to the licensee that the NRC is not satisfied with the licensee's performance (5).

c. The integrated assessment process would not provide a measure of how good licensee performance was. This was due in part to the significant resources involved and the lack of clear guidance against which good performance can be measured. Therefore, performance issues involving solely good or neutral licensee performance would not be included in the evaluation. **To what extent and how should positive inspection findings be factored into an assessment process?**

General comments

The vast majority of respondents stated that positive inspection findings should not be factored into the assessment process. One respondent stated that positive inspection findings should be factored into an assessment process when they provide insight into the context for negative findings. NEI responded that neither positive or negative inspection findings should be graded or weighted in the assessment process. There was one dissenting opinion.

Other views

The NRC should use positive inspection findings in the assessment process. Failure to consider positive findings in those areas would tend to skew the outcome in the negative direction. The assessment process should treat all of the observations that are reported in a balanced manner (9).

d. The integrated assessment process would include an assessment report for each licensee and a public meeting with the licensee to review this assessment. **How should the NRC's assessment results be communicated to the licensees and to the public?**

General comments

There was a wide variety of responses. The industry recommended that the NRC make PIs publicly available on a quarterly basis. Because performance indicators are traceable and subject to scrutiny, lengthy communication between the licensee and the NRC would not be necessary. Other respondents suggested various types of processes that involve periodic public meetings. One respondent suggested that the assessment report be published in the *Federal Register* several weeks prior to a public meeting.

Other views

The assessments results of a specific plant were provided to the public relative only to that plant's prior performance. It would be more meaningful for the public to hear about plant performance in context with performance at other plants. Annually, The NRC Regional Administrator should brief the Commission at a public meeting of the performance of all plants within his or her region. The Regional Administrator should also outline regulatory priorities for each plant in the upcoming year (3).

e. The integrated assessment process would provide several opportunities for the licensee and the public to be made aware of the issues being considered and to provide feedback and input on these issues and assessment results. **What are the most desirable ways to include licensee and public input and feedback during the implementation of the assessment process?**

General comments

The vast majority of respondents stated that feedback from licensees and the public during the assessment process was not warranted . Some respondents supported the use of public input on proposed corrective actions and enhancements to the assessment process. There was one dissenting opinion.

Other views

Licensee feedback (other than times where significant findings result) should be a regular part of communications that take place between the licensees and NRC resident inspectors and project managers. Public feedback should be solicited through the use of forms made available at public document rooms for regular ongoing feedback. If significant findings result from the assessment process, public meetings should be held to provide the public with a forum for airing concerns and obtaining information (1).

There are already ample opportunities for the licensee input into the process. However, the role of the public in the assessment process should be enhanced. This should include release of preliminary assessment findings to the local public document room, public meetings to discuss the progress and findings of the assessment, and a mechanism to gather public input and answer questions about the assessment (5).

Feedback from the licensees and the public should not be factors in the assessment process for individual plants. However, the NRC should consider feedback from licensees and the public on enhancements to the assessment process (3).

2. Assessment Criteria

a. In the integrated assessment process, a plant performance matrix is used to categorize performance findings into assessment areas in order to provide better structure for the information and to better communicate assessment results. **What additional or alternate information should be used and how should it be integrated?**

General comments

There was a wide variety of responses. Some respondents supported the proposed matrix, others supported the matrix with modifications, and the industry didn't support the proposal at all.

Other views

There are some significant weaknesses in the information being used in the staff proposal. The NRC should focus on objective safety performance and not on individual errors (7).

This information must also define how long the problem existed prior to detection and how many reasonable opportunities for discovery were missed in order to determine relative risk and the effectiveness of the licensee's self-assessment process (3).

NRC should not use a plant performance matrix or similar concepts that rely upon a grading of performance in functional areas or cross-functional areas. This would lead the public to believe that plants with relatively low grades would be unsafe. Additionally, assigning numeric grades would cause the process to be subjective and not tied directly to safety (10).

Additional information is not necessarily needed for the plant performance matrix to work. The regulator must have the will and freedom to act upon their findings (13).

The proposed assessment process, as described in SECY-98-045 using the PIM template provides an excellent basis to structure the assessment finding and communicate results (5).

b. Under the integrated assessment process, individual performance issues were numerically graded on the basis of safety and regulatory significance. As stated in the SRM for SECY-98-045 dated June 30, 1998, the Commission did not approve of this approach. **Are there alternate methods by which the NRC could provide a quantitative input into the assessment process so that the significance of issues can be assigned in a scrutable way?**

General comments

The industry did not support this approach and presented its own proposal as outlined in the NEI white paper. Public advocacy groups supported the proposal made in SECY-98-045 except for one respondent who stated that the staff's proposal would be premature at this stage in the evolutionary process.

c. In developing a new assessment process, it was essential that the results of the assessment could be clearly communicated to the licensees and the public. The staff chose color category ratings for each assessment area for the integrated assessment process. As stated in the SRM for SECY-98-045 dated June 30, 1998, the Commission did not approve of this approach. **What alternate presentations could be used to clearly convey the results of licensee performance assessments?**

General comments

There was a wide variety of responses and no true consensus. The industry stated that the staff's approach is flawed because the scores and numbers are much too subjective and not a true indication of safety performance. Two respondents suggested that quantitative results could be illustrated as a trend plot and that qualitative results should be communicated as being inside or outside of NRC criteria. One respondent supported the staff's proposal.

Other views

Results of the assessment need to be clearly communicated to the licensee and the public using measurable means against standardized criteria. Performance should be assessed on the basis of safety such as indications of an increase in safety, adequate safety, or a reduction in safety (2).

The assessment scheme should place plants in the same color, grade, category, or rating when they have comparable performance based on objective criteria (3).

3. Decision Model

The staff developed a decision model to provide for a structured and predictable application of NRC actions in response to assessment results. **Are there additional or better ways to optimize the scrutability and predictability of the NRC outcomes of the assessment process?**

General comments

The majority of respondents stated that the decision model was ineffective because it is too subjective and retains the flaws of the existing process. One respondent generally supported the concept of the proposed decision model.

Other views

The outcome from the IRAP process was not a decision model that afforded structured and predictable application of NRC actions. The outcome of the IRAP process was a proposed model that retained the fundamental flaws of the existing process (3).

4. Assessment Periodicity

The staff recommended that an annual performance assessment be performed for each plant to allow for a periodic assessment report and a public meeting to discuss the assessment results. **Is there a more appropriate periodicity for accurately assessing changes in licensee performance?**

General comments

There were a wide variety of responses. The industry suggested the use of quarterly PI's; others suggested a floating schedule tied to performance or completion of a set of specified inspections; some agreed with the proposed annual assessment.

Other views

Assessments should be performed on a continual basis using a rolling time frame as a reference (1).

The assessment frequency should be based on the completion of a specified set of inspections which may not necessarily be on an annual time frame (2).

Rather than set a fixed schedule for assessment such as one year, the assessment frequency should be tied to trends in regulatory performance but in no case should exceed two years (5).

An annual assessment would be fine. However, the NRC should take the initiative to openly address declining licensee performance as it occurs (13)

5. Success Criteria

a. The integrated assessment process was designed to produce NRC assessments that are more scrutable and predictable. **For comparison, how scrutable, predictable, and objective are the current assessment processes?**

General comments

There was strong agreement from all respondents that the SALP process and watch list were not scrutable, objective, or predictable.

b. The integrated assessment process was intended to be less resource intensive for both the NRC and the licensee. **How do the estimated licensee costs compare with the costs of the existing assessment processes?**

General comments

Only two responses were received to this question. One respondent stated that the process does not sharpen the safety focus of inspection activity, or minimize reporting requirements. The other respondent stated that licensee costs are not an appropriate basis for judging an assessment process.

C. Risk-Informed Assessment Guidance

36. Effective risk management is necessary to ensure the safe operation of nuclear power plants. **How should indications of risk-management performance be considered in the assessment of plant safety?**

General comments

Several industry representatives stated that risk insights gleaned from Individual Plant Evaluations and the maintenance rule can be used to identify risk important plant indicators and to set thresholds for performance. Further, it was stated that providing adequate protection to the public health and safety requires assessment of three areas: (1) the events which can challenge plant safety systems; (2) the mitigation systems; and (3) the three barriers to radionuclide release.

A related comment from a public interest group stated that risk information should continue to be used to define the thresholds for *reporting* problems (emphasis added). Risk information should not be used to discard or discount reported problems.

Other views

In the absence of a rule to mandate risk assessment at facilities, any NRC attempt to use "indications of risk-management performance" will fail. ELPC recommends that risk assessment be mandated and that the NRC use "indications of risk-management performance" as a factor in determining regulatory oversight (5).

If the NRC applied risk insights to discard or discount LERs and other data it collects on plant performance, it might reach the very undesirable condition of basing plant assessments, and therefore its inspection priorities, on a paucity of information (3).

Indications of risk management performance should be given the highest consideration in that good risk-management has the highest potential for ensuring public health and safety (1).

Risk assessment is based upon the premise that all reactors meet their licensing basis and thus have a certain level of defense in depth. As the Millstone experience and the hundreds of DER's indicate, this is simply not the case (13).

2. One aspect of a risk-informed regulatory process is that plant performance measures are considered commensurate with their impact on plant safety and risk. **Are the questions presented in "Guidance for Assessing the Risk Inherent in Plant Performance" sufficient to ensure that inspection findings are interpreted in a risk-informed manner?**

General comments

Several industry representatives stated that: (1) inspection findings do not measure plant safety performance, they measure compliance with the regulations; (2) inspection findings, whether interpreted in light of risk or not, should not be used to form the basis for an integrated assessment process; and (3) the integrated plant assessment process should focus on those aspects important for protecting public health and safety (i.e., the events which can challenge plant safety systems, mitigation systems, and the three barriers to radionuclide release)

Consistent with that perspective, another respondent stated that in place of the staff's proposed guidance, NRC's assessment process should focus on performance criteria that reflect risk-informed considerations. It was stated that, as an example, station blackout tends to be an important factor in risk assessments for many plants, so NRC may want to consider establishing performance criteria for use in its assessment process that are tied to the reliability and availability of the emergency diesel generators. It was stated that establishment of performance criteria could occur as part of a more broad-based set of performance criteria that are tied to the performance goals established under the Maintenance Rule.

A public interest group stated that the questions and their associated potential evidence should not be used in their present form because they are vague, ill-defined measures.

Other views

Most of the questions in the guidance document can be better answered by looking directly at performance indicators of safety performance. Engineering and design is an exception, which does not lend itself to objective safety outcome indicators. These areas can best be measured through audits, self assessments, and inspections. The areas of human performance and problem identification and resolution are really only germane if performance is below the regulatory thresholds of objective safety performance indicators. If performance is above the thresholds, the performance in these areas is by definition acceptable (7).

I think that the key to assuring that the information is used appropriately is to have NRC staff personnel with the experience necessary to understand and apply risk insights. As part of the NRC's PRA implementation plan, the use of Senior Reactor Analysts needs to be expanded and their insights applied on a consistent basis (1).

In general, the guidance identifies factors involving performance that might affect risk, but does not actually call for determination of risk in judging performance (10).

1. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," presents a framework, principles, and staff expectations relative to regulatory decision-making.

   a. **What role, if any, should the guidance of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis" play in risk-informed assessments of plant performance?**

      <u>General comments</u>

      Several industry representatives stated that this guidance can be used in establishing safety thresholds for risk-significant systems, structures, and components (e.g., the safety threshold for mitigating systems performance could be set at a level that corresponds to a level that could cause a change in conditional core damage probability that would require NRC approval).

      Response from one public interest group stated that: (1) the guidance in the Regulatory Guide is sound; (2) the Guide should be part of implementation of a Commission rule on risk: (3) it can, nonetheless, provide guidance for risk-informed assessment of plant performance; and (4) if used properly, the Guide should be capable of ranking non-compliances with the regulations according to their importance to protecting health and safety.

      On the other hand, another public interest group stated that the framework, principles, and staff expectations expressed in Regulatory Guide 1.174 cannot be practically used in the assessment process. Further, it is stated that the current state-of-the-art of PRA methodologies makes it impractical to assign weights to factors in the assessment process based on PRA results.

Other views

It should play the same role as any other regulatory guidance. It provides clarification to the NRC's goals and objectives in meeting current regulations (1).

b. **What role should PRA techniques and risk metrics play in the assessment of plant performance?**

General comments

A public interest group stated that PRA techniques and risk metrics should be used to establish thresholds for reporting problems, but that PRA techniques and risk metrics should not be used to rank problems reported when risk-informed thresholds are exceeded.

Several industry representatives stated that: (1) PRA and risk insights should be used as much as practicable in order to orient the integrated assessment toward what we know about risk; (2) risk insights should be used to determine what indicators and thresholds to apply; and (3) the need is recognized for defense in depth, both in barriers to radionuclide release and in the methods used to mitigate accidents.

One comment stated that the thresholds could differ from plant to plant depending on risk identified by the plant-specific PRA.

4. **How should patterns of degrading human performance, equipment performance, and risk management at a nuclear power plant be factored into the plant performance assessment process?**

General comments

Several industry representatives stated that: (1) if human or equipment performance is degrading such that it affects safety performance, then it will show up in the safety performance indicators; (2) if performance falls below the regulatory threshold, the cause may be human performance, or equipment degradation, or some other cause, which should be evaluated by the plant corrective action program; and (3) if performance falls below the regulatory threshold, NRC should increase its focus and look more closely.

A public interest group stated that: (1) the proper NRC response to an apparent problem area is to further examine the reasons for the problems; and (2) if examinations reveal programmatic breakdowns, the NRC must initiate actions that cause these breakdowns to be corrected.

Other views

The NRC's primary responsibility is to ensure that its licensees provide adequate protection to the public health and safety. The licensees are responsible for managing their people, processes, procedures, and plant equipment to achieve safe and reliable nuclear power (7).

With respect to risk-informed assessment guidance, it is assumed that the problem reporting thresholds will be effective in keeping non-relevant information out of the process. For example, human errors involving non-power block issues and problems involving only non-safety related equipment should not enter the process and therefore do not need special treatment within the process (3).

Again, risk management has no place in the assessment of nuclear reactor performance. How has NRC accounted for the fact that Haddam Neck's ECCS was undersized for 28 years or that a valve on the borated water tank at Big Rock was broken for 13 years? Until these deficiencies are addressed by your regulators and accounted for in your PRAs, the industry should not be allowed to use risk insights as a means of avoiding regulation (13).

5. **Are the questions raised in "Guidance for Assessing the Risk Inherent in Plant Performance" sufficient to provide a risk-informed assessment of plant safety that addresses the influence of human performance and equipment performance on plant safety?**

General comments

A public interest group stated that the questions and their potential evidence should not be used in their present form because they are vague, ill-defined measures.

Industry representatives stated that: (1) the questions are not sufficient to provide a risk-informed assessment of plant safety, but the questions will help the NRC staff score and bin issues in its matrix; (2) a risk-informed assessment of plant safety requires a set of objective safety performance indicators which relate to protection of the public health and safety; (3) equipment performance will be apparent in the safety indicators; and (4) human performance, as it is important to plant safety, will also be reflected in the safety indicators.

Other views

Acceptable guidance should take the form of the explanations and examples provided in NUREG-1022 on reporting requirements under 10 CFR 50.72/50.73 (3).

D. INDICATORS

1. General

The trending methodologies can be used as part of an integrated assessment process that uses both quantitative and qualitative information. The trending methodologies are not intended to be used in isolation as the only definitive identifying element in plant performance assessment.

a. **How should the NRC use quantitative measures of performance?**

b. **What methodologies and/or performance measures would be useful to quantitatively monitor plant performance trends?**

General comments

The industry stated that the NRC should base its assessment on objective indicators that directly measure safety performance and risk- informed thresholds, and that the current NRC indicators and trending methodology do not do this. Several respondents suggested that NRC use NEI's indicators or similar indicators. One public interest group stated that the problem is not the indicators, but failure to act upon the performance measures.

2. Trending Methodology

a. The staff considered more than 20 variables during the development of both the trend and the regression models.
   1. **Are there other variables that should be considered?**
   2. **Are the data for the suggested variables publicly available?**
   3. **Are the data for the suggested variables reported to the NRC?**
   4. **How frequently are the data for the suggested variables available (e.g., daily, weekly, quarterly, annually, etc.)?**

b. The staff considered a variety of time periods for monitoring plant performance during the development of the trend model. The proposed trend model uses a four-quarter moving average. **Should a different time period be used?**

c. The proposed trend model uses a "hit" threshold that is based on a fixed 2-year average of one standard deviation beyond the quarterly industry mean for the period from July 1995 through June 1997. **Should a different threshold be used?**

d. The proposed trend model uses a discussion candidate threshold value of two hits. **Should a different threshold be used?**

General comments

The industry stated that variables used by the NRC are not indicators of safety performance, are duplicative, are subjective, and do not directly affect plant safety. The industry's proposed indicators provide a more direct indicator of safety and a trend of safety performance. The industry stated that hits and thresholds should be based on risk insights and should not be based on relative performance, and that a threshold based on 1995-1997 performance is inappropriate. Other public interest group comments expressed concerns with the current variables.

Other views

Longstanding design problems, such as the multiple ice condenser problems at DC Cook are not accurately reflected in the safety system reliability variables (3).

Some licensees are not properly reporting problems; some licensees do not report degraded conditions identified by NRC inspectors; therefore the LER database is not an accurate source of data on nuclear plant problems (3).

3. Financial Variable Trend Methodology

a. Financial indicators can be used to gain insight into licensee performance in conjunction with other assessment measures. They would not be relied upon solely to draw conclusions on licensee performance in an integrated assessment process. **How should financial indicators be used in the assessment of licensee performance?**

General comments

The majority of the 11 respondents to this question stated that the financial indicators should not be used. In addition, the majority of the respondents stated the following reasons given for not using financial indicators: (1) the NRC role should be to assure financial qualification and adequate decommissioning funding, (2) the staff should not be looking for indicators that would have predicted discussion at previous senior management meetings, (3) the purpose of indicators should be to determine how well licensees are performing in protecting public health and safety, and (4) financial indicators are statistical correlations that do not prove causality.

Other views

Given competitive pressures on the utilities and the potential for significant reductions in the financial resources available to support nuclear plant operations, the NRC should consider expanding its review of plant performance to include examination of financial indicators (5).

Financial performance indicators may be useful in determining why a licensee's performance has suffered. However, the regulator should be able to determine performance problems prior to evincing themselves in financial indicators (13).

Only use as an "additional consideration" because there is no relevant data that shows a direct relationship between financial performance and plant risk that is not better measured by safety measures (1).

b. **Are there any financial methodology processes that will provide a more useful set of financial variables?**

The majority of the seven respondents to this question stated that this was an issue for utility management and the financial community. In addition, the respondents stated that financial indicators are not a predictor of safety outcomes or plant safety. The respondents also stated that the financial indicators would penalize plants that shut down to correct problems and could cause inappropriate behavior to satisfy NRC indicators.

Other views

Use the amounts of civil penalties as a financial variable (3).

c. The financial variables are based on publicly available data. Are there other financial data that could be made available that would be more useful?

General comments

The majority of the six respondents to this question stated that no other financial data could be made available that would be more useful than publicly available data and that it would not be in the utilities' interest to expand financial data they provide to the public. The respondents noted that at least one utility is presently providing less information to FERC to protect its competitiveness.

Other general comments

The following comments were taken from selected comment submittals, because although they do not apply directly to any specific question, they presented generic statements applicable to the proposed assessment process.

We strongly support the NRC's goal of developing a new method of assessing licensee performance at commercial power plants, as long as those new methods complement, rather than replace, enforcement of strict compliance with existing safety requirements (5).

We believe that the framework developed at the workshop and the consensus reached on a key set of fundamental assessment issues provide a far superior approach to plant assessment than the process referenced in the Federal Register (7).

In our work involving the most troubled plants the following items became obvious to my coworkers and me: (1) management failings were the root cause of all significant problems, (2) management failings among middle management and lower tier supervision were an outcome of management failings at the highest levels of the organization, (3) senior management failings generally involved inadequate provision of resources to Operations, Maintenance, Training, Engineering, or all of the above, (4) senior management failings took place 2 to 8 years before the assessment process discerned significant safety problems of a magnitude to result in plant shutdown, (5) a tedious and expensive corrective action process would be required before the plants would again be fully operational, (6) the assessment process does not recognize and correct senior management failings at their onset (28).

Attachment 7a

List of respondents

1) Terry J. Herrmann (concerned citizen)
2) Ohio Dept of Public Safety
3) Union of Concerned Scientists
4) Florida Power Corp.
5) Environmental Law and Policy Center
6) Southern Company
7) Nuclear Energy Institute
8) Entergy Operations, Inc.
9) Licensing Support Services
10) Morgan, Lewis & Bockius LLP
11) Charles R. Jones (consultant/concerned citizen)
12) Commonwealth Edison
13) Public Citizen's Critical Mass Energy Project
14) Northern States Power Co.
15) TU Electric
16) Arizona Public Service Co.
17) Florida Power and Light Company
18) PECO Energy
19) Northeast Utilities Co.
20) Southern California Edison
21) Tennessee Valley Authority
22) South Carolina Electric and Gas Co.
23) Entergy Operations Inc. (duplicate)
24) Wolf Creek Operating Corporation
25) Pacific Gas and Electric Company
26) Public Citizen (duplicate)
27) New York Power Authority
28) Dolphin Enterprises (consultant)

COMMITMENTS

Attachment 8

1/4/99

Response to SECY-98-045 SRM and SRM M981102

The SECY-98-045 SRM dated June 30, 1998, and SRM M981102, issued in response to the November 2, 1998, Commission briefing on reactor oversight process improvements identified 12 specific areas of Commission interest. The areas and how they are addressed in this Commission paper are summarized below:

SECY-98-045 SRM

4. **Enforcement should not be used as a "driving force" of the assessment process.**

   The staff removed from the assessment process the use of enforcement results to assign significance to inspection findings. The new performance assessment process is not driven by enforcement.

5. **The staff should continue to include positive findings in inspection reports.**

   Positive inspection findings will remain in inspection reports. In addition, the staff will strive to provide proper context for all inspection findings.

6. **The assessment process should not be based primarily on a scoring of plant issues matrix issues at this time.**

   The new assessment process is based on a balanced use of PIs and cornerstone inspection areas. Developmental efforts to assign significance to inspection findings in a risk-informed manner are still in progress.

7. **Efforts devoted to developing leading indicators should be commensurate with the probability of success. The Commission supports the development of indicators that can identify emerging safety problems (in a leading [if possible], or at least a concurrent manner).**

   The staff has worked with the industry to identify a set of PIs for near term implementation. When used within the assessment process, events or conditions that make up these indicators are considered leading indicators to unsafe plant operation. Some of the PIs are considered leading to other PIs (e.g., transients lead to trips). These are described in Attachments 1 and 2. The staff is continuing efforts to develop risk-based PIs for use in the future.

8. **The definition for performance rating categories should not be "color coded."**

   Performance category ratings are not proposed for the new performance assessment process. The only use of color terminology is to describe thresholds for PIs and cornerstone inspection areas in a shorthand fashion. The thresholds are also given word identifiers (for example, the white threshold is the interface between the licensee response band and the regulatory response band).

9. **The staff should remain open to less dramatic change which might integrate the existing processes in a manner which saves resources and may be more readily**

implemented. The process should address regional consistency and sensitivity of the process to varying levels of inspection effort. Conceptual changes to the inspection program needed to conform with the new assessment process should also be identified.

The new regulatory oversight process framework that was developed with significant inputs from internal and external stakeholders is actually more dramatic than the concept proposed by IRAP. In particular, significant changes are being proposed not only to assessment, but also to inspection and enforcement. Licensee involvement will increase in terms of generating and submitting voluntary PI data.

Regional consistency will be improved through the increased use of objective performance indicators, increasing the focus on identifying the significance of inspection findings, and more clearly defining action decisions in an action matrix. In addition, program office reviews of process implementation will be continued.

The new assessment process is designed to accommodate varying levels of inspection effort in several ways. PIs, which are independent of inspection effort, constitute a significant portion of the assessment process input. Inspection findings with very little significance, which are highly dependent on level of inspection effort, are governed so that they do not have a significant impact on assessment results. Risk-significant findings, however, are less dependent on level of inspection effort, and based on their significance, will be used directly for assessment.

Significant inspection program changes, such as defining a new risk-informed baseline inspection program, are described in Attachment 3.

<u>SRM M981102</u>

4. **Refine key definitions**.

   Key terms are defined in Attachments 2, 3, and 4.

5. **Identify attributes that are important to the assessment program but are not covered by performance indicators**.

   Areas requiring inspection because they are not adequately covered by PIs are identified in Attachments 2 and 3. These areas are identified as cornerstone inspection areas in the proposed assessment process, Attachment 4.

6. **Identify the different types of information that would be used in the assessment process and the methodology that would be used for deriving an objective and scrutable overall assessment of licensee performance.**

   The new assessment process uses a balance of PIs and inspection findings grouped by cornerstone area. Objectivity and scrutability are obtained by clearly defining thresholds for the assessment inputs. Section 3.2 of Attachment 4 describes the process for conducting the assessment.

7. **Identify the desired outcomes of the "cornerstones," particularly related to the capabilities of mitigation systems and design barriers to perform as intended. Identify the process for implementing regulatory action.**

   The desired outcomes of the cornerstones are reflected in the thresholds that were developed for each PI (described in Attachment 2) and will be developed for cornerstone inspection areas (described in Attachments 3 and 4).

   Regulatory action decisions will be triggered in response to assessment inputs crossing thresholds. These decisions will be governed by the action matrix shown in Attachments 1 and 4.

8. **Define the proposed vehicles to inform the Commission of assessment results and to inform the public.**

   Attachment 4 describes the processes that will be used to communicate assessment results. The Commission will have negative consent approval of all assessment results and actions. An annual Commission briefing will be held to present the results of all assessments. In addition, individual meetings with licensees will be held shortly after the Commission meeting to discuss assessment results and planned actions.

9. **Provide the methodology the staff will use to verify and validate the efficacy of the improved oversight process.**

   Benchmarking efforts, a 6-month pilot program, and ongoing evaluations of regulatory oversight process improvement effectiveness are described in the transition plan section of this Commission paper.