April 19, 2006

The Honorable George V. Voinovich
Chairman, Subcommittee on Clean Air,
Climate Change, and Nuclear Safety
Committee on Environment and Public Works
United States Senate
Washington, D.C. 20510

Dear Mr. Chairman:

I am pleased to provide the enclosed responses to the post-hearing questions that were submitted by members of the Committee from the March 9, 2006 hearing to conduct oversight on the Nuclear Regulatory Commission.

If I can be of further assistance, please do not hesitate to contact me.

Sincerely,

/RA/

Nils J. Diaz

Enclosure: As stated

cc: The Honorable Thomas R. Carper

Response to Post Oversight Hearing Questions

QUESTIONS FROM SENATOR INHOFE:

1. As you know, for many years, I have been advocating that a stable and predictable licensing process is an absolute must if we are to proceed with constructing new nuclear plants in this country. In fact, we changed the law in the Energy Policy Act of 1992 to address the problem of a utility having to get a Construction Permit and then an Operating License. Although we have made numerous changes to improve this process and a number of utilities have already begun developing their application for Combined Operating License, the Commission is still tinkering with the rule on the licensing process. During the hearing, the Commission stated that the Part 52 rule will not be finalized until mid-January. What do you suggest to those utilities that are currently in the middle of developing their applications?

   ANSWER:

   The NRC's proposed changes to the Part 52 requirements governing the contents of combined license applications are not significantly different from the current requirements. The majority of the changes involve clarification and reorganization of the existing requirements and the addition of requirements to address operational program information (e.g., information on programs such as occupational dose control, physical security, and fitness for duty) to implement recent Commission policy decisions in this area. The Commission believes that revising Part 52 at this time, on the brink of a potential renaissance of nuclear energy in this country, will provide long term benefits, not only for future license applicants, but for prospective applicants who may be developing applications before the final rule is issued. The proposed requirements provide a greater level of specificity than the current requirements and therefore should be a useful aid to companies that are currently preparing combined license applications. The amendments to the rule clarify the applicability of various requirements to each of the licensing processes addressed in Part 52.

   Since the proposed rule is publically available, applicants and other stakeholders have access to the changes being considered. In addition, on March 14, 2006, the NRC staff conducted a public workshop to facilitate stakeholder comments on the proposed rule. During this workshop, the NRC staff discussed the proposed changes for the Part 52 requirements and answered stakeholder questions on these changes. The NRC believes that this workshop clarified further the bases for the proposed changes and should also aid companies preparing combined license applications.

2. I agree completely that it is the responsibility of each applicant to submit a complete and quality application that meets all of the NRC's requirements and guidance. Having said that, I understand that the nuclear industry has been working for several years with the NRC and is currently in its fifth round of revisions to develop guidelines on what a "complete and quality" application entails. When do you expect this regulatory guidance to be finalized? Also, what steps are you taking to ensure that your Standard Review Plan is developed to match the application guideline? When will your Standard Review Plan be available?

Enclosure

ANSWER:

NRC guidance for new reactor applications includes a planned combined license (COL) application regulatory guide (DG-1145) for use by applicants preparing COL applications and an update of pertinent standard review plan (SRP) sections for use by NRC staff reviewing COL applications.

The NRC staff estimates that the COL application regulatory guide will be completed by December 2006, which is compatible with the schedule for the promulgation of the final Part 52 rule. The NRC staff is scheduled to issue the draft COL application regulatory guide (DG-1145) in June 2006. In the interim, the staff is placing draft work-in-progress sections of DG-1145 on the NRC web site to solicit early stakeholder feedback and interaction. Several public workshops have been scheduled to discuss these draft work-in-progress sections as they become available. There will also be one or more public workshops after DG-1145 is issued formally for public comment. This COL application regulatory guide contains the information that COL applicants need to provide in their applications. The schedule for issuing DG-1145 supports prospective applicants who are planning to submit COL applications in late 2007 and 2008.

Complementary to the COL application regulatory guide, the NRC staff has developed an SRP update plan to support the anticipated new site and reactor licensing applications. The SRP and the plan for updating the SRP is publicly available on the NRC web site at: www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/. The staff has prioritized the SRP section updates to support new reactor licensing activities and expects to complete all of the high priority SRP Section updates by December 2007. The high priority sections are those most important to reviewing COL applications. The medium priority SRP sections represent knowledge transfer and have been previously used during the design certification review process. The staff has identified a goal of having 70 percent of medium priority sections updated by December 2007, with the balance of these sections scheduled for completion in 2008. The low priority SRP sections were either recently updated or are of low safety significance for new reactor designs.

The NRC staff has taken several actions to ensure consistency between these documents. The project management staff that has the responsibility for managing the development and updates to both documents has been reorganized so that they are in the same part of the organization. In addition, the same technical staff will support both guidance documents. Furthermore, the NRC staff plans to leverage the development of the COL application regulatory guide during the subsequent updates to the SRP. Specifically, during public comment solicitations, the NRC staff plans to extend the scope of these interactions to include both the COL application regulatory guide as well as the associated SRP sections. This will allow early public interaction on the SRP update.

3. One of the problems that caused the licensing process to bog down in the late 1970s and the 1980s was that there was no end to reopening issues during the licensing process. What steps is the NRC taking to ensure that legitimate safety and technical issues get resolved promptly, as required by your safety mandate, but that once they are resolved, they do not get reopened?

ANSWER:

Differences between the new licensing processes and those in place in the late 1970s and early 1980s are expected to clarify and address questions about relitigation. Under the 10 C.F.R. Part 50 licensing process, a construction permit (CP) could be issued on the basis of preliminary design information. This preliminary design was often changed and finalized during plant construction, and the modified design was submitted in an operating license application that was filed after the CP. Thus, when a CP holder subsequently requested an operating license (OL) during construction, it would be the first time that the staff reviewed the new final information and it would be the first opportunity for the new information to be challenged in a hearing. Construction-related quality assurance problems also contributed to protracted litigation, including reconsideration of matters previously thought to have been resolved. In addition, the hearing process – including the scope of contentions being raised – was affected by the need for the NRC and the industry to address, on a generic basis, a number of significant safety and technical issues, notably issues stemming from the Three Mile Island Unit 2 accident in March 1979.

The current licensing regime in 10 C.F.R. Part 52, which the Commission first established in 1989, provides for an applicant to submit more detailed, final design information to support a combined license (COL) application. The licensing process in Part 52 is itself intended to provide the opportunity to resolve certain important issues early in the process -- through the standard design/design certification and early site permit processes -- before the Commission issues a COL and plant construction begins. While litigation is by its nature somewhat uncertain, the Commission believes Part 52 offers a much more stable and predictable licensing process than that previously used.

The Commission recently issued a proposed rule to amend Part 52 to explicitly clarify many procedural matters, thus reducing the risk of litigation of such matters in the first COL proceedings. Of course, even under Part 52, late design changes by an applicant, a poor-quality application, or construction-related issues could still cause delays apart from the hearing process.

With respect to the procedures for reopening a matter previously litigated before the Atomic Safety and Licensing Board (ASLB) or raising a late-filed contention, the NRC's procedural regulations ensure that the NRC complies with the Atomic Energy Act of 1954, as amended, and the Administrative Procedure Act, and does so with due regard for the rights of all parties and without undue delay of the adjudicatory proceeding. These regulations, which were most recently revised in 2004 to streamline the agency's Rules of Practice, build upon earlier changes to ensure that adjudications focus on matters that are truly material to the NRC's licensing decisions and that decisions are reached in a timely manner. These actions should further reduce the potential for unnecessary delay in NRC proceedings from matters raised late in licensing proceedings.

4. As I mentioned in my opening statement, I am concerned about the length of time it takes to issue new, risk-related regulations. What can you do to address this problem?

ANSWER:

The NRC has completed several activities that involve issuing new risk-informed regulations: changes to the requirements for the design of combustible gas control systems (10 C.F.R. 50.44); adoption of performance-based fire protection design requirements from the national fire protection code (10 C.F.R. 50.48); and risk-informed categorization and special treatment requirements for structures, systems, and components (10 C.F.R. 50.69). In addition, the NRC recently published a proposed rule to provide a voluntary, risk-informed design basis for the emergency core cooling system design (10 C.F.R. 50.46a). Most recently, the Commission has approved issuance of an advance notice of proposed rulemaking to solicit stakeholder comments about broader risk-informed revisions of our reactor regulations particularly as they relate to new reactor designs.

The time period associated with completion of these regulations included both the time to develop a sound technical basis to demonstrate that there would be reasonable assurance that public health and safety will be adequately protected when risk-informed changes in the regulations are implemented and the time to implement the rulemaking process to ensure that stakeholder comments are solicited and considered and that other statutory obligations are met. Risk-informed regulatory requirements must also be supported by sound analysis methods, and the NRC has worked closely with national consensus standard bodies on appropriate standards to ensure consistent treatment of the analysis methods and results.

Nevertheless, the Commission will continue to pursue ways to improve the timeliness and effectiveness of the rulemaking process and closely monitor the progress of the staff's activities associated with risk-informed changes to the regulations, as well as the other high priority rulemaking activities to codify new security requirements and improve the licensing process for new reactors.

QUESTIONS FROM CHAIRMAN VOINOVICH:

1. New Plant Licensing:

In your testimony, you stated that the Commission may receive 11 or more applications for new nuclear plants in the next few years, beginning in 2007. At the same time, NRC will have to review two Design Certification applications for new reactor designs.

a) How many NRC staff (or FTEs) is needed in FY 2007 and FY 2008 to deal with this workload without delays?

ANSWER:

Since the passage of the Energy Policy Act of 2005, industry has accelerated its plans to submit combined license (COL) and early site permit (ESP) applications. This has created challenges to the staff's ability to sustain the planned review schedules of 24 months for an ESP, 30 months for a COL application, and 42 months for a design certification. To sustain originally planned review schedules based on current industry plans, the NRC estimates that, after accounting for expected attrition and the currently identified new work associated with new reactor licensing, it will need to hire approximately 350 to 400 new staff each year for the next two to three years. This estimate is subject to change primarily because the industry continues to make additional announcements about its plans for future reactor applications.

b) Does the Commission's proposed FY 2007 budget reflect the preparatory work necessary for receiving 11 or more combined license (COL) applications?

ANSWER:

No. At the time the FY 2007 budget was developed, the NRC was expecting 4 COL applications for new nuclear power reactors. Based on currently available information, the nuclear industry plans to submit 13 to 15 combined license applications to NRC during 2007 - 2009. This has affected the staff's ability to sustain the planned review schedules described in Question 1.a above. The staff estimates that in order to sustain the planned review schedules, an additional $40 million in budget authority will likely be required for FY 2007 to support the additional preparatory activities and pre-application consultations for the expected COL applications. The NRC is taking steps to address the need for additional resources for the expanded workload.

c) Has the Commission considered more staff to these projects as a way to gain scheduling efficiencies?

ANSWER:

The NRC currently projects the need for additional resources by FY 2008 as described in response to Question 1.a and b. to support the currently identified new work associated with new reactor licensing. The level of effort associated with these resources is consistent with our design-centered approach, which will use a single technical decision to support multiple combined license applications for the same technical area of review when common elements of multiple applications permit. This should significantly improve the efficiency of our process, but this assumption is predicated upon applicants providing a consistent level of standardization of the applications.

As to increased efficiency with even more staff than what is described above, some incremental improvements may be possible, but these are unlikely to improve dramatically the already aggressive proposed schedules.

d) Has the Commission devised a specific training program to get the new employees qualified to work on these applications? If yes, then please describe it for the Committee.

ANSWER:

The agency recognizes that the challenge to assimilate new employees fully into the agency goes beyond ensuring that they possess the technical expertise needed to make a safety determination. To be assimilated fully into the agency, new employees will need to be oriented and trained in our regulatory processes as well as our internal business processes. The NRC maintains a wide range of engineering and regulatory courses, which are conducted at the agency's Technical Training Center and Professional Development Center. When a new employee reports for the first day of work, the Office of Human Resources provides an employee orientation from an agency-wide perspective. In addition, the Office of Nuclear Reactor Regulation (NRR), which will experience the largest growth in new employees, has developed a new employee orientation and training program, which includes training courses, seminars, self-study activities, and partnering with more experienced employees, to expedite the new employee's adjustment to the office and need for training. Training topics include regulatory processes, such as the licensing process and allegations, and, business processes, such as information security, computer security, Freedom of Information Act, and the agency electronic document storage system. Additional position-specific qualification or training plans are developed for employees to gain detailed knowledge in specific areas of expertise, such as new reactor licensing, risk assessment, or health physics. The Office of Human Resources, in partnership with NRR, is scheduling additional courses, as needed, to support the orientation and training program and the staff training plans. Further, to ensure that the agency provides the right training for the influx of staff, a job task analysis is currently being conducted to identify needed skills and knowledge areas explicitly for new reactor licensing. From this assessment, the agency can develop new courses to prepare the large number of employees needed to meet the projected demand for new reactor licensing activities.

Additionally, the NRR new employee orientation and training program explicitly requires a new employee to be paired with a more experienced employee from the new employee's immediate working group that will serve as a guide to help the new employee learn the business processes of the agency. This is especially important to assimilate new employees due to the challenge of office space. Because of the lack of office space, new employees can not be co-located with their working group. In addition, mentors and subject matter experts will continue to be utilized as resources for new employees to discuss technical and regulatory issues.

By using a combination of training, self-study, seminars, and current employees as mentors, the agency is confident that new employees will be trained and assimilated into our organization.

e) Delays in the licensing process make nuclear power a less attractive investment to utilities and Wall Street. What is the NRC doing to reduce the risk of delay in the licensing process?

ANSWER:

The 10 C.F.R. Part 52 regulations promote stability by prescribing the various licensing processes for the NRC and applicants to govern the issuance of early site permits (ESPs), standard design certifications, and combined licenses. Part 52 provides for resolution of important issues at an early stage in the licensing process, prior to applicants expending significant resources on plant design and construction. Furthermore, both the NRC and applicants learned valuable lessons during the review of the three first-of-a-kind ESP reviews with respect to resolving issues, such as the need for the applicant's early interactions with State and local officials, proposal of new methodologies (for example, seismic), and the tracking and timely resolution of public comments. As noted in our response to Senator Inhofe's Question 1, the staff is currently pursuing proposed revisions to 10 C.F.R. Part 52 to enhance the NRC's regulatory effectiveness and efficiency in implementing its new reactor licensing and approval processes.

The NRC staff has developed a practical and efficient "design-centered licensing review approach" for the review of the reactor technology and the COL applications that reference the technology. The objective of the design-centered licensing review approach, which is, in effect, a complement to the design certification process in Part 52, is to conduct one technical review for each reactor design and use this one decision to support the design certification and multiple COL applications. Successful implementation of such an approach depends upon applicants choosing a consistent level of standardization. The overall quality and completeness of an application can have a substantial impact on the review schedule; therefore, as noted in our response to Senator Inhofe's Question 2, the staff is working to develop a COL Regulatory Guide and to update the SRP guidance.

With respect to the conduct of fair, orderly, and efficient hearings, the Commission amended its Rules of Practice for adjudications in 2004 to include model milestones for the conduct of contested proceedings, including hearings held in connection with ESP and COL applications. See 10 C.F.R. Part 2, Appendix B. The regulations provide that the presiding officer should use the milestones as a starting point and set detailed litigation schedules based upon all relevant information.

f) Currently, the NRC estimates a design certification process could take as long as 60 months to complete. Could a multinational design approval program (MDAP) allow the NRC to shorten the schedule for completion of design certifications? What are the resource and budgetary implications of MDAP in terms of costs and benefits?

<u>ANSWER</u>:

It is too early to say how much the MDAP will affect the schedule for NRC review of future design certification applications. The benefits of the MDAP on the NRC's schedule to complete design certification reviews depends on many factors. The first factor is the degree of similarity among the designs proposed in the U.S. and internationally. The second contingent factor is the level of review undertaken by the participating regulatory agencies to meet their national standards and how similar these standards are to those of the NRC. In order for the NRC to shorten its overall design certification schedule, it would be critical to shorten its review in technical areas. Some areas of technical review will have more benefits than others due to design or regulatory differences. The MDAP will provide the NRC additional information regarding potential technical issues during pre-application reviews and allow the agency to leverage work done by our foreign regulatory peers (e.g., work on AREVA's Evolutionary Power Reactor (EPR)) prior to the actual submittal of applications. The staff will consider the benefits of international cooperation, particularly the technical information that can be leveraged from other regulatory agencies, when developing a review schedule for a reactor design that is also being reviewed by our international counterparts. For the coming generation of U.S. power reactors, the MDAP will benefit the NRC's safety review of the EPR design. Other designs that prospective applicants have already chosen (Westinghouse's AP1000 and General Electric's ESBWR) are not currently expected to be reviewed by the NRC for the U.S. market utilizing a multinational approach.

The MDAP will require a small amount of resources (about 2 FTE in FY 2007) to plan and coordinate interactions with the NRC's foreign counterparts. It is anticipated that these resources could be offset by improved effectiveness and efficiency of the NRC's design certification process because the NRC will be alerted to various issues and insights encountered by its MDAP partners in their design certification reviews.

g) I am encouraged by the NRC's plan for a new "design-centered approach" to help move applications along by allowing common issues for the three new reactor types to be resolved generically. To what extent will this approach speed up the schedule for licensing a new plant? Are there any legislative changes needed to help establish an expedited licensing process?

ANSWER:

The staff's objective in the design-centered licensing review approach is to conduct one technical review for each reactor design and use the resulting decision to support the design certification and multiple COL applications. The design-centered licensing review approach optimizes the review process for the large number of anticipated new reactor licensing applications while providing quality technical and safety reviews in accordance with the NRC's regulations. The staff estimates that utilizing this approach could result in approximately a 40% savings in FTE and a 35% reduction in schedule as compared to that necessary to perform individual design certification and combined license reviews. Success of the design-centered licensing review approach depends on industry's willingness and ability to standardize COL applications referencing the same reactor design.

At this time, the staff has not identified a need for any legislative changes to support an expedited licensing process.

2. Human Capital and Infrastructure Challenges:

Your budget proposal for FY 2007 projects staffing at 3,309 employees.

a) What is your best current projection for total FTEs at the NRC for the next five years? To the extent possible, please explain the projected increases/decreases in the aggregate and by function including new reactor licensing, Yucca Mountain licensing, nuclear security, license renewal, power uprate application and others that are appropriate.

ANSWER:

Based on current information, NRC expects the FTE for most of its programs to remain relatively constant over the next five years. However, NRC expects a net increase of 500 to 700 FTE over the next five years to review new reactor licensing applications, DOE's license application for the Yucca Mountain high-level waste repository, industry applications to increase the number of fuel cycle production facilities, and potential NRC involvement in other initiatives. This would result in a total FTE projection of 3,700 - 3,900 for the next five years. This projection is based on current information and is subject to change.

b) What steps are you taking to train and assimilate new hires into your organization? Is there a formal training/qualification program to ensure that they understand the formal regulatory processes used by the NRC?

ANSWER:

As described in detail in response to Question 1.d., the NRC maintains a wide range of engineering and regulatory courses, which are conducted at the agency's Technical Training Center and Professional Development Center.

c) Presumably, the majority of new employees that you are bringing on board to replace the retiring employees are recent college graduates with little or no relevant work experience. What is the NRC doing to compensate for the inevitable "brain drain"?

ANSWER:

NRC is using a variety of human capital strategies to maintain its technical knowledge and skills during a time when experienced staff members are increasingly eligible to retire and current and new employees need the benefit of their knowledge. These include the use of authorities NRC obtained in the Energy Policy Act of 2005 to waive dual compensation limitations for re-hired annuitants with critical skills, offers of retention allowances to keep highly-skilled technical staff members on board, and knowledge management tools and techniques.

The NRC is recruiting a mixture of recent graduates and experienced professionals. Approximately 25% of NRR's FY 2006 new hires are entry level (i.e., recent college graduates). The remaining 75% are experienced professionals (some with an excess of 20 years experience) from nuclear generating companies, architect-engineering firms, consultants, military, etc. Therefore, our training/development and knowledge management programs consider the needs not only of current employees but of both entry-level and experienced new hires, particularly the "what we do" and "how we do it" information unique to NRC's safety and security mission.

Knowledge management is a top priority at the NRC and we are working to better integrate initiatives in this area across the agency. The NRC is implementing an agency-wide knowledge management program that is designed to provide an overarching framework for the agency.

Examples of NRC knowledge management (KM) tools and techniques in place or being tested for broader application include:

- knowledge capture interviewing of experienced staff
- cataloguing expert document collections
- establishing electronic communication groups for a network of people, centered on critical business practices, who come together virtually to share and learn from others experiences, insights, and best practices
- maintaining a KM website with best practices, tools, conferences and seminars, and other information
- identifying KM champions and staff leads for each office and region to facilitate choosing and implementing appropriate KM tools and techniques.

d) I am encouraged by the agency's ongoing effort to institutionalize the lessons learned as mentioned in your testimony. I think this is absolutely necessary considering that hundreds of new people that you are bringing onboard may not have even heard of "the Davis-Besse incident" for which the agency went through such an extensive corrective action program. When do you expect to complete this program so that new employees will benefit from a collection of corporate knowledge?

ANSWER:

The base Lessons Learned Program is expected to be implemented in June 2006. This base program consists of a Lessons Learned Program Management Directive, the required implementing procedures, and assignment of staff to implement the program. The base program will apply to new lessons learned going forward. It is anticipated that the full program, which includes web-based staff and public access to a growing record of historical agency lessons learned information, will be completed in fall 2007.

e) I understand that NRC has a goal of hiring 350 people annually for the next several years, and as a result, the agency will need additional office space to support this growth. During the hearing, you and other Commissioners stated that the agency may need support from this Committee in working with the General Services Administration in acquiring additional space in close proximity to the agency's Rockville campus. Please explain the situation and how the Committee can help.

ANSWER:

The NRC's accelerated hiring program will steadily exhaust the space in our headquarters building, despite our aggressive space optimization program. The NRC is working with GSA in a two step approach to address the growth associated with new work. The Commission wrote a letter to the Committee on April 5, 2006, which provided details on the NRC's office space requirements and how the Committee can assist the NRC with obtaining appropriate space. A copy of the letter is attached (Attachment 1).

f) During the hearing, I mentioned middle management as one of the problems in the Federal government that I have observed from my other committee chairmanship (Subcommittee on Oversight of Government Management, the Federal Workforce and the District of Columbia). We do not do a very good job of bringing people in from outside, who can bring different ideas and approaches to problem solving. How is the agency doing in this regard?

ANSWER:

The NRC has historically filled the majority of its supervisory and managerial positions from within the agency, placing graduates of two highly competitive programs, the Leadership Potential Program (LPP) for movement into first- and second-line supervisory positions and the Senior Executive Service (SES) Candidate Development Program (CDP) for movement into the SES. This is especially true in cases where the first-line supervisor is expected to have technical skills and knowledge as well as provide administrative oversight and leadership. Thus, the technical experience gained at the NRC is deemed to be extremely valuable in the selection process for supervisory positions. In the administrative offices, however, there is a more diverse mix of managers who were selected from both within and outside the agency.

The NRC hires almost exclusively from the outside for its full performance level scientific and engineering positions. These hires feed the pipeline for the staffing of first-line supervisory positions, which in turn feeds the applicant pipeline for middle management positions. Therefore, the external experience and ideas these hires bring to the NRC serve to make our agency more diverse at all levels. Currently, approximately 20% of the agency's supervisors and managers have less than ten years of service with the NRC.

3. Implementation of the Energy Policy Act of 2005 Provisions:

a) The NRC has taken measures on radioactive materials licenses, through orders and rulemaking changes, to enhance the security of radioactive materials in quantities of concern. There must be a coordinated effort in the regulation of radioactive materials security. How does NRC intend to address this need in its ongoing effort to regulate materials security? Does the NRC plan to expand on its current enhanced security requirements to include Category 3 and other materials?

ANSWER:

After the 9/11 terrorist attacks, the NRC initiated a comprehensive security assessment of its licensees, including radioactive materials licensees, to determine whether additional security measures were warranted. This effort identified a number of immediate, intermediate, and long-term actions needed to enhance the security of risk-significant radioactive materials in an elevated threat environment. These actions have included issuing safeguards advisories, issuing Orders to licensees imposing additional security and control requirements, conducting rulemaking to establish new security requirements and to incorporate the requirements imposed through Orders into NRC's regulations, and developing a National Source Tracking System. These actions are based on a "graded" approach; in general, licensees possessing significant quantities of radioactive material or material that is potentially more attractive to adversaries require more rigorous security measures to be in place. This effort has involved and been coordinated with other Federal agencies, including intelligence and law enforcement agencies; State regulatory agencies; NRC licensees and industry groups; and international partners, such as Canada, Mexico, and the International Atomic Energy Agency (IAEA).

The NRC will continue to address the potential need for additional or revised security measures through a risk-informed and integrated approach that also includes an evaluation of the adequacy of existing regulations against the threat environment. The ongoing efforts will continue to be coordinated with other Federal agencies, State agencies, NRC licensees and industry groups, and international partners. Consistent with the IAEA Code of Conduct on the Safety and Security of Radioactive Sources, the NRC's efforts have focused on materials licensees possessing or authorized to possess Category 1 and Category 2 sources. The NRC, as part of the National Source Tracking System rulemaking, solicited and received stakeholder comments on the need to track Category 3 Sources. The issue of whether or not to include Category 3 sources as part of the National Source Tracking System is currently under consideration by the Commission. The NRC is also evaluating its existing programs as they apply to sources below Category 2 quantities to identify areas where increased licensee accountability or access control requirements may be warranted for Category 3 sources.

b) The NRC has announced and asked for public comment on their plans to establish a Radiation Source Protection and Security Task Force, with the NRC as its chair, to evaluate and provide recommendations relating to the security of radiation sources in the United States. Is the NRC planning to involve individuals and organizations outside of the government into this task force?

<u>ANSWER</u>:

Yes. A representative of the Organization of Agreement States and the Conference on Radiation Control Program Directors is a non-voting member of the Task Force. Each Subgroup of the Task Force also has a non-voting member representing State interests. In addition, the Task Force sponsored a closed facilitated stakeholder meeting with representatives of State and local government organizations.

4. <u>Reactor Oversight Process (ROP)</u>:

During the hearing, you testified that the revised ROP, which was implemented in April 2000, has matured and improved. Separately, during our private meeting on January 30, you mentioned that there has been a significant improvement in overall safety at nuclear power plants as demonstrated by the number of plant events, shutdowns, and extended shutdowns in the last few years. Please quantify for the Committee this improvement. Additionally, does the Commission believe there is a correlation between the improved safety records at nuclear plants and the implementation of the ROP? Do recent trends in inspection findings and performance indicators support your conclusion?

ANSWER:

The NRC initiated an Industry Trends Program (ITP) to monitor trends in indicators of industry performance as a means to confirm that the safety of operating power plants is being maintained. Should any indicators show a statistically significant adverse trend, the NRC evaluates them and takes appropriate regulatory action using its existing processes for resolving generic issues and issuing generic communications. The NRC formally reviews these indicators each year, and any adverse trends are reported to Congress in the NRC's Performance and Accountability Report. No statistically significant adverse trends have been identified to date.

Over the past ten years, most of the ITP indicators show improved operating performance. The latest results can be found on the NRC web site at: www.nrc.gov/reactors/operating/oversight/industry-trends.html, and are attached (Attachment 2) for your convenience. Also, attached (Attachment 3) is a chart that shows an annual count of shutdown months resulting from unplanned extended reactor shutdowns. This chart was first presented at the 26th Annual INPO CEO Conference and shows a significant improvement in industry performance in this area.

The NRC believes that its Reactor Oversight Process (ROP) has had a positive impact on improving nuclear plant safety in the United States. All commercial nuclear power plants in the United States are inspected as part of the ROP. All plants receive a baseline level of inspection activity independent of their overall performance. When significant performance problems are identified, the NRC performs additional inspections to ensure that the licensee takes appropriate corrective actions. In addition, the NRC inspects on a graded approach to determine if other plant problems exist. If significant problems exist, the NRC has the regulatory authority to either confirm that certain actions be taken or issue Orders for certain actions, which could include a plant shutdown. This approach to regulatory oversight is risk-informed and performance-based, which allows the NRC to focus resources on weaker performing plants.

Over the past six years of implementation, there have been a total of 5,529 inspection findings identified by the licensees the NRC regulates and by NRC inspectors through routine inspection activities. These findings can be accessed by the public on the NRC's public web page. Of this total number of inspection findings, the NRC has identified over 4,239 findings, some of which were of moderate to high safety importance. The remaining 1,290 findings were identified by the licensees. Although the NRC has not quantified this impact on the improved safety record of commercial nuclear plants, the NRC is confident that these inspection findings correlate to improved safety. The overall effectiveness of the ROP is reported each year to the Commission.

5. Public Confidence:

During the hearing, I emphasized the importance of the NRC's redoubling its efforts to shore up public confidence. Chairman Diaz briefly summarized the Congressional district office outreach program as an example of the NRC's recent public relations efforts. Please describe the NRC's current public relations programs so that the Committee can better assess the agency's efforts in this very important area.

ANSWER:

The NRC conducts a number of programs and initiatives for bolstering public confidence and employing openness as a key cornerstone in agency communications and its regulatory processes. Building and maintaining public trust is critical to carrying out the NRC's mission. Recognizing that openness must be balanced with national security concerns, the NRC employs a strategy to make as much information available to the public without providing information that would be useful to potential terrorists.

Our strategic plan identifies the following key strategies to support openness:

- Provide accurate and timely information to the public about the uses of and risks associated with radioactive materials.

- Enhance the awareness of the NRC's independent role in protecting public health and safety and the environment.

- Provide accurate and timely information to the public about the safety performance of the licensees regulated by the NRC.

- Provide a fair and timely process to allow public involvement in NRC decision-making in matters not involving sensitive unclassified, safeguards, classified, or proprietary information; provide authorized and cleared individuals security information as needed.

- Obtain early public involvement on issues most likely to generate substantial interest and promote two-way communication to enhance public confidence in the NRC's regulatory processes.

Examples of how we carry out these strategies include holding an annual public meeting near each nuclear power plant site to discuss the plant's safety performance and NRC's oversight of the plant. We maintain an up-to-date website with user-friendly information that is of interest to the public and provide a direct portal to the vast majority of NRC documents that are public. This information helps the public understand agency decisions and to participate effectively in the regulatory process. Additionally, we engage the public early in rulemakings and reactor license renewals, explaining the process in public meetings near a plant before an application is discussed. Numerous public meetings and workshops are held each year to obtain input on key issues such as new reactors, emergency preparedness, and high-level waste. We also develop communication plans for high-profile issues and agency decisions to communicate with a wide array of stakeholders, including Congress, the news media, licensees, Federal, State and local governments, the general public, and the international community. We also use fact sheets and brochures to convey information.

NRC's Office of Public Affairs has a forward-leaning approach to communicating the NRC's message to reporters and responds rapidly to the media to correct misstatements. We also use "For the Record" on our website to post accurate information on issues in the media. In addition to issuing news releases frequently, we talk on a regular basis to reporters who cover NRC activities to ensure that the reporters have accurate information on the activities they are reporting on. Use of op-eds, and press releases in advance of meetings is also helpful in communicating directly with citizens.

Interviews, speeches and press conferences by key agency officials is another way to get our message out. When Commissioners travel to a licensed facility or to another country, they typically talk to the media in the area about their visit and other issues of interest. In addition, all public Commission meetings are broadcast live over the internet and are available in the archives for viewing at any time by those who are interested. We are also exploring the possibility of providing podcasts, a method of publishing files to the internet so that they can be downloaded, of other public meetings that reporters and the public can access from our website.

Recently, we developed a web page to be used during emergencies involving licensed facilities to keep the public informed of our actions to keep them safe. Plans are also underway to produce a new video about the NRC that, in conjunction with our DVD on security at licensed nuclear facilities, will be offered to educational and vocational institutions around the country and be available on our website.

As you know, the offices of Public Affairs, Congressional Affairs, and State and Tribal Programs have collaborated on a local Congressional outreach program that meets with representatives of selected Congressional district offices and local government officials. During this pilot program, the NRC has been meeting with Congressional staff across the country in the members' home States. The purpose of the outreach program is to ensure that Congressional offices are kept aware of NRC activities and the status of nearby licensed facilities.

6. Organizational Performance and Efficiency:

During the hearing, one of the management issues I highlighted for the Commission was the need to apply the "Total Quality Management" concept to continually improve the agency's performance and productivity. The NRC has to be more efficient in order to meet the unprecedented challenges associated with the anticipated workload. Please describe the Commission's effort to improve the organizational performance and efficiency. What metrics do you have in place to assure you are making progress in this area and what feedback have you received from stakeholders?

ANSWER:

Regarding performance and productivity, each year since FY 2002, the NRC has met all targets established for the agency's Strategic Plan safety and security goals. The NRC is committed to ensuring that resources are well managed. Productivity and output measures have been met except in cases where safety and security concerns took precedence. During this period, at least two key process improvements have been achieved in each of the Reactors, Materials, and Waste programs.

Annually, the Commission provides guidance on the agency's outcome-based performance measures, which indicate the level of success needed to achieve the agency's goals. In addition, the NRC identifies which activities support the NRC's outcome-based performance measures and uses these as guides to formulate the budget. Beginning in FY 2006, the NRC has developed a number of efficiency measures for the activities under the agency's two major program areas of Nuclear Reactor Safety and Nuclear Materials and Waste Safety. The measures support the agency's Strategic Plan goal of Efficiency and Effectiveness as reported in the agency's FY 2007 Performance Budget to Congress. As examples, the NRC intends to achieve an average five percent reduction in license renewal resources for applications in FY 2007. The NRC also plans to implement process enhancements to permit a five percent improvement in the timeliness of acting on rulemaking petitions. Further, the agency plans to reduce resources expended in support of incident response and emergency preparedness exercises by five percent while still accomplishing agency goals for each exercise. In addition, the enforcement process for handling discrimination allegations has targeted a 10 percent reduction in the average enforcement processing time.

With respect to efficiency measures, a number of examples include FY 2004 gains associated with the Reactor Oversight Process (ROP), which resulted in approximately fifteen FTE savings per year for reactor inspection activities. Further, improved implementation guidance in a license renewal regulatory guide and standard review plan has resulted in a permanent thirty percent efficiency gain in resources needed to review license renewal applications. In addition, since 2000, materials licensing labor rates have been reduced twenty-two percent for new applications, ten percent for amendments, and fifty percent for renewals. During the same period, materials inspection labor rates have been reduced by thirteen percent while the number of inspections required was reduced by twenty-four percent. These efficiencies were achieved without impacting program performance.

During the NRC's most recent Program Assessment Rating Tool (PART) review, the Office of Management and Budget (OMB) recognized the validity of such measures as supporting long term efficiency gains. Additionally, since 2003, five of NRC's seven major programs dealing with Nuclear Reactor Safety and Nuclear Materials and Waste Safety have been subjected to OMB's PART review screening. Of the programs evaluated, four were rated as "effective," which is OMB's highest rating, with the fifth rated as "moderately effective," the second highest rating. An important component to receiving favorable PART ratings is the adoption and use of effective performance and efficiency measures to gauge the results of the programs.

With respect to stakeholder feedback, the NRC has been favorably evaluated by OMB and a number of other stakeholders. As examples, in 2004, the NRC staff conducted a survey with stakeholders that measured the effectiveness of NRC's strategic goal of enhancing openness in our regulatory process. The survey mainly involved local and county officials living near nuclear plants because they are opinion leaders who could influence residents in their surrounding communities and because they already may have some knowledge of the NRC and our regulatory activities. The NRC's overall

"Satisfaction" score was 68 out of 100, which is relatively high for a regulatory agency, particularly for the first measurement. The survey results showed that NRC staff was found to be professional, competent, and helpful. We also received high scores for the information we provide to our stakeholders. The respondents seemed to be more satisfied with the openness of NRC than the opportunities the agency offers them to participate in the regulatory process. The 2004 government-wide scores, including those for the NRC, were subsequently published in *The Washington Post*.

As a follow-up, the Commission will conduct a series of focus groups to help identify how the NRC can improve these messages and ultimately enhance confidence in the regulatory process, enable the NRC to assess more specifically how much the public knows about the NRC, and determine their perceptions about nuclear security, emergency planning and safety issues. The results of the focus group effort, combined with the results of the survey on "openness," will give us a better awareness of the specific elements of public outreach that need to be enhanced and the next steps required of our public outreach efforts.

7. Nuclear Security:

In your testimony, you stated that the Commission is making good progress in implementing the security provisions that this Committee passed as part of the Energy Policy Act of 2005, such as a rulemaking on the revised Design Basis Threat. However, I want to make sure that after the rulemaking is completed NRC does not continue to require security changes without going through the appropriate process. According to a report (OIG-05-A-19) from the NRC Inspector General, the NRC has issued a series of safeguards advisories (total of 65) from September 11, 2001 to January 26, 2005. The OIG determined that 40 advisories, out of 65, were used for requesting or requiring information or licensee action, containing regulatory guidance, and conveying apparent requirements, without going through the established process required by the Administrative Procedures Act. What steps has the NRC taken to respond to the concerns identified in that report?

ANSWER:

The NRC has implemented steps to include the Paperwork Reduction Act provisions and to include a specific consideration of potential backfit impacts on licensees. Additionally, the NRC now also includes a statement in its Advisories that affirms the specific Advisory contains no new requirements. Further, the NRC is incorporating security Advisories into the established generic communications process, which will formalize the process of issuing safeguards advisories and embody the salient provisions of the Administrative Procedure Act.

8. Research and Test Reactors:

It is my understanding that the Massachusetts Institute of Technology had submitted an application for a power uprate of its research and test reactor in 2001, but the NRC has yet to act on it. What is the current status of the agency's review of this application? When do you expect to complete the review?

ANSWER:

The Massachusetts Institute of Technology (MIT) requested a power uprate of 20% (5 MW thermal to 6 MW thermal). MIT submitted the request as part of its license renewal application. The NRC staff has treated it as one action because of the interdependence of the analyses and review. After the initial review of the application, NRC sent three requests for additional information (RAI) to the licensee in 2001 and 2002. The licensee responded to all three RAIs by letter on January 29, 2004. Since 2004, the staff has focused efforts on other competing priorities, including security and other license renewal reviews of RTRs whose licenses would expire before MIT's.

The staff is resuming the review of the adequacy of the licensee's response to the RAIs to determine whether there is an adequate technical basis to approve the amendment for the power uprate and license renewal. The goal for completion of this safety evaluation and amendment to the license is the middle of FY 2007.

9. USEC:

USEC's planned American Centrifuge Plant (ACP) will be located on the DOE Portsmouth reservation, will utilize the GCEP buildings constructed by DOE, and will use centrifuge technology developed by DOE. There are on-going DOE remediation efforts throughout that site and DOE is also constructing a DUF6 conversion facility adjacent to the planned ACP. I understand that DOE will lease the GCEP buildings to USEC under an amendment to the existing lease for the enrichment site. It is also my understanding that DOE has concluded that it is appropriate to continue the DOE Price Anderson indemnification for the ACP. Does NRC agree with DOE's decision to continue the DOE Price Anderson indemnification of those areas leased for the ACP?

ANSWER:

The NRC has not concluded its analysis and consideration of legal issues regarding the insurance requirements for the proposed USEC ACP facility. At this time, NRC is, as part of its analysis, discussing this matter with DOE.

QUESTIONS FROM SENATOR JEFFORDS:

1. You state in your written testimony that the NRC has approved 108 power uprates to date, with approximately 17 more applications pending. How much power is that exactly, and what was the regulatory cost associated with the application review and other NRC actions that were necessary to get that power?

ANSWER:

The 108 power uprates to date represent about 4599 megawatts-electric, or the equivalent of about 4.6 large nuclear power plants.

There are three types of power uprates. Measurement uncertain recapture power uprates are less than 2 percent and are based on the use of enhanced techniques for calculating reactor power. Stretch power uprates are typically up to 7 percent and are within the design capacity of the plant. Extended power uprates are greater than stretch power uprates, have been approved for increases as high as 20 percent, and require significant plant modification. The 108 power uprates approved by the NRC since 1977 include 34 measurement uncertainty recapture power uprates, 60 stretch power uprates, and 14 extended power uprates. Since 1998, the average regulatory cost associated with the review of the three types of power uprate applications has been as follows: (1) a measurement uncertainty recapture power uprate application has used, on average, about 0.7 FTE of staff review effort; (2) a stretch power uprate application has used, on average, about 1.3 FTE of staff review effort; and (3) an extended power uprate application has used, on average, about 3.1 FTE of staff review effort. Currently, 7 applications for 10 plants are under review.

2. Several organizations argue that the Independent Safety Assessment (ISA) that was done at Maine Yankee in 1996 is the "gold standard" of plant inspections. They say this because of the length of time it took, and because of the systems that were examined at the plant. I understand that after the NRC's experience at Maine Yankee it changed its inspection procedures to incorporate lessons learned from that experience and to focus inspections on safety issues. Am I correct in my understanding that since 1996 the NRC now focuses more inspection attention on plants with known safety problems?

ANSWER:

You are correct, the NRC focuses more inspection attention on plants with known safety problems. As part of the development of the Reactor Oversight Process (ROP), the NRC used lessons learned from the 1996 Maine Yankee inspection as well as other lessons learned reports and information. The current regulatory framework for the ROP is a risk-informed, tiered approach to ensure plant safety. There are three key strategic performance areas: reactor safety, radiation safety, and safeguards. Reviews of plant performance, using both the performance indicators and inspection findings, determine what additional action the NRC will take if there are signs of declining performance. The process utilizes different levels of regulatory response with NRC oversight increasing as plant performance declines. As performance declines, additional NRC resources are applied with inspection teams focused on the cause of issues and overall degraded performance.

3. Is it also correct that the Maine Yankee suffered from an inspection deficit which is why a team of 24 people were needed to do the ISA?

ANSWER:

No. The NRC Chairman at the time directed the independent safety assessment (ISA) in response to concerns about safety and regulatory oversight associated with the emergency core cooling system analyses. As stated in the Maine Yankee ISA report,

> In December 1995, the Union of Concerned Scientists forwarded anonymous allegations to the State of Maine, and the State submitted the allegations to the NRC. The allegations were that Yankee Atomic Electric Company knowingly

performed inadequate analyses to support an increase in the rated thermal power at which Maine Yankee Atomic Power Station (MYAPS) may operate. After performing a technical review, the NRC Office of Nuclear Reactor Regulation (NRR) issued a confirmatory order on January 3, 1996, limiting power operation at the plant to the original licensed power level of 2440 MWt.

The NRC Office of the Inspector General (OIG) completed an inquiry into this allegation on May 8, 1996. OIG established that MYAPS had experienced problems with, and made modifications to, the RELAP/5YA computer code which was used in the emergency core cooling analysis for a small-break loss-of-coolant accident. The problems and subsequent modifications were not reported to the NRC as is required and the code was not used in accordance with the Safety Evaluation Report and with the Three Mile Island Action Plan Item II.K.3.3.1. OIG also reported weaknesses in the NRC review and followup activities which contributed to NRC failure to detect these deficiencies.

The RELAP issue raised a question of whether similar problems existed in other areas. In order to address this question, as well as to respond to concerns by the Governor of Maine about the safety and effectiveness of regulatory oversight at Maine Yankee, the NRC Chairman initiated an independent safety assessment of MYAPS. This assessment was to be performed by a team comprised of staff who were independent of any recent or significant regulatory oversight responsibility for Maine Yankee. Additionally, the assessment was to be coordinated with the State of Maine to facilitate participation by State representatives consistent with the Commission's policy on cooperation with States at commercial nuclear power plants.

4. Will you provide the Committee with a document that lists the systems, procedures, and particular equipment inspected at Maine Yankee in 1996 during the Independent Safety Assessment and in 2004 during the independent engineering assessment at Vermont Yankee?

ANSWER:

The October 7, 1996, Maine Yankee report listed the overall goals of the independent safety assessment. The goals were to: (1) independently assess the conformance of MYAPS to its design and licensing bases including appropriate reviews at the site and corporate offices; (2) independently assess operational safety performance giving risk perspectives where appropriate; (3) evaluate the effectiveness of licensee self-assessments, corrective actions, and improvement plans; and (4) determine the root cause(s) of safety-significant findings and draw conclusions on overall performance.

An in-depth assessment was conducted in the areas of plant operations, maintenance, testing, engineering, analytic code support, and self-assessment and corrective actions. The assessment consisted of interviews; system walkdowns; extended control room observations; system reviews of service water, high pressure safety injection, and emergency diesel generators; program, process, and procedure reviews; and analytic code reviews. In addition, an extensive reliability analysis of auxiliary feedwater, emergency feedwater, high pressure injection, and emergency diesel generator systems was performed. Emphasis was placed on identifying both licensee strengths and performance weaknesses. The press release issued on October 8, 1996, that summarizes the ISA findings is attached (Attachment 4).

In selecting samples for the Vermont Yankee review, the team focused on the most risk-significant components and operator actions. The team selected these components and operator actions by using the risk information contained in the licensee's Probabilistic Risk Assessment (PRA) and the NRC's Simplified Plant Analysis Risk (SPAR) models. Many of the samples selected were located within the reactor core isolation cooling, main feedwater, safety relief valve, onsite electrical power, and off-site electrical power systems. In addition, inspection samples were added based upon operational experience reviews.

A complete listing of all components, operator actions and operating experience issues reviewed by the inspection team is contained in Attachment A to the Vermont Yankee report (Attachment 5). A total of 91 samples were chosen for the team's initial review. Based on a number of considerations, 45 of the original 91 samples were selected for a more detailed review. The staff used Temporary Instruction 2515/158, "Functional Review of Low Margin/Risk Significant Components and Human Actions," to conduct this inspection.

5. Constituents have also argued that the Independent Safety Inspection done at Maine Yankee in 1996 should be repeated at other plants because it was independent of the NRC. Constituents liken it to having an outside audit of a plant. My understanding is that the inspectors that did the inspection were independent of the plant and of the region, but only few were contractors. Most were NRC employees. Is that correct?

<u>ANSWER</u>:

Most of the inspectors were NRC employees. The ISA team members were independent of the NRC Region I office, the Office of Nuclear Reactor Regulation (NRR), and the plant. The Independent Safety Assessment (ISA) team comprised 25 members: 16 NRC members, 3 State of Maine members, and 6 contractors. The team was organized with five functional area leaders reporting to a team leader. The team leader reported to the team manager, who reported directly to the NRC Chairman.

6. Senator Clinton has asked the NRC to conduct an Independent Safety Assessment at the Indian Point plant in her state. As you know, a similar request was made by citizen groups during the power uprate process at Vermont Yankee. The Advisory Committee on Reactor Safeguards determined that this level of inspection was not needed at Vermont Yankee in order to determine the power uprate could proceed. My understanding of your commitment during the hearing to Senator Clinton is that the NRC will conduct an engineering inspection at Indian Point, similar to that done at Vermont Yankee during the power uprate. Is my understanding accurate? Will you provide me with a copy of the letter you agreed to send Senator Clinton during the hearing summarizing the inspection commitment you announced for Indian Point?

ANSWER:

Yes, your understanding is correct. NRC will conduct an engineering team inspection (similar to the inspections conducted at Vermont Yankee) at each Indian Point unit. The inspection will be focused on the review of plant components significant to safety; the inspection is expected to last seven weeks, including four weeks of on-site time and approximately 700 hours of direct inspection. The inspection for Unit 2 is scheduled to begin in January 2007, and Unit 3 in September 2007. The inspection will include an evaluation of changes to the plant's licensing basis to ensure that safety margins remain adequate.

The letter to Senator Clinton is attached (Attachment 6). We received a subsequent letter from Senator Clinton, dated April 3, 2006, and are developing a response.

7. When the Independent Safety Assessment was conducted at Maine Yankee in 1996, legislation was not required. The NRC had sufficient legal authority to conduct such an inspection. Several House members have introduced legislation to require an Independent Safety Assessment at Indian Point. Does the NRC now need legal authority to conduct such an inspection? Does the NRC support plant-specific legislation to set inspection protocols?

ANSWER:

The NRC currently has the legal authority to conduct inspections at nuclear power plants. Plant-specific legislation is not necessary to set inspection protocols.

8. I understand that NRC conducted the inspection at Maine Yankee because, in December 1995, anonymous allegations were sent to the State of Maine and to the NRC regarding falsification of computer modeling in the plant's power uprate analysis. The plant's power had been boosted in 1989, several years earlier. It was alleged that the NRC staff knew the modeling was faulty, and colluded with the plant owners to conceal that fact. The NRC Inspector General did an investigation. The NRC Chairman at the time, in response to the IG report, and a request from the Governor of the State of Maine, ordered the Independent Safety inspection. Are you aware of any possible criminal activity or collusion between NRC staff and the operators at Indian Point over modeling or any other aspect of plant operation?

ANSWER:

The Commission is not aware of any criminal activity or collusion between NRC staff and the operators at Indian Point over modeling or other aspects of plant operation.

9. I also want to ask a question about the scope of the Maine Yankee Independent Safety Assessment. I have also been told that this was a superior inspection because it was a thorough top to bottom look at the plant's operation. My understanding is that it was an in-depth look at some safety systems, but not an entire audit in the popular sense. The inspectors did not look at the entire plant, and they did not look at external issues, such as emergency evacuation plans. They did not examine every nut and bolt and every piece of paper. Is that correct?

ANSWER:

That is correct, the independent safety assessment did not include external plant issues, such as, emergency preparedness. As stated in the Maine Yankee report, "An indepth assessment was conducted in the areas of plant operations, maintenance, testing, engineering, analytic code support, and self-assessment and corrective actions. The assessment consisted of interviews; system walkdowns; extended control room observations; system reviews of service water, high pressure safety injection, and emergency diesel generators; program, process, and procedure reviews; and analytic code reviews. In addition, an extensive reliability analysis of auxiliary feedwater, emergency feedwater, high pressure injection, and emergency diesel generator systems was performed. Emphasis was placed on identifying both licensee strengths and performance weaknesses."

10. There are repeated calls among New Englanders to revive a 10 year old inspection procedure that was used once. What can be done to give the public more confidence in NRC's current inspections, and particularly the inspections of older plants that may have changes to their license conditions?

ANSWER:

All commercial nuclear power plants in the United States are inspected on a continual basis as part of the Reactor Oversight Process (ROP). All plants receive an annual baseline level of inspection activity independent of their overall performance. When significant performance problems are identified, the NRC performs additional inspections to ensure that the licensee takes appropriate corrective actions. In addition, the NRC inspects on a graded approach to determine if other plant problems exist. If significant problems exist, the NRC has the regulatory authority to either confirm that certain actions be taken or to issue Orders for certain actions, which could include a plant shutdown. This approach to regulatory oversight is risk-informed and performance-based, which allows the NRC to focus resources effectively on weaker performing plants.

The ROP is very open to the public in that all inspection procedures, inspection reports, and assessments are available through the NRC's public web page. Public meetings are held to discuss certain important inspection findings and on an annual basis, overall assessment of licensee performance is discussed with the licensee in a meeting open to the public.

When a licensee elects to amend its license, the licensee must submit a license amendment application to the NRC. This is a formal process that involves a high degree of regulatory review, including whether the proposed change to the license is safe from a public health and safety perspective, in addition to an assessment of environmental effects. The results of these assessments are also publically available.

The 10-year old inspection that was conducted at Maine Yankee referred to in this question is not part of the NRC's regulatory oversight process as described above. However, important elements of this inspection can be performed by the NRC when licensee performance has resulted in significant performance deficiencies. Currently, there are no plants in the Northeast United States whose performance demands such an inspection. In addition, the NRC conducts a rigorous team inspection referred to as the Component Design Bases Inspection. This inspection examines the structures, systems, and components at each plant to confirm that important selected components will perform as they are intended to prevent serious accidents. All plants in the Northeast will receive this inspection within the next two years.

The NRC believes that these processes effectively protect public health and safety and the environment and are open to the public.

11. The NRC recently released a draft rule on the design basis threat for public comment. In the Energy Policy Act of 2005, Congress clearly directs NRC to consider 12 factors as part of the DBT rulemaking, including the need to defend against attacks by large groups, attacks by air, and other types of attacks. Instead, the Commission has chosen not to address 6 of the 12 factors as directed by Congress and has solicited public comment on "whether or how" all 12 matters should be addressed. I am concerned that deferring the analysis to the final rule is contrary to the rulemaking process, because it makes genuine comment impossible. How does the NRC legally justify its decision not to examine in its draft all of the 12 factors identified in the Energy Policy Act of 2005?

ANSWER:

Section 651(a) of the Energy Policy Act directed that while the NRC is conducting its rulemaking to revise the Design Basis Threat (DBT) set forth in its regulations, it shall "consider," along with other factors, twelve factors specified in that provision of the Act. The NRC did consider each of the factors in developing the text of the proposed rule. In addition, the Federal Register notice (FRN) of proposed rulemaking (70 FR 67380) enumerates all twelve factors and asks for comments on whether or how each of the factors should be addressed in the rule (70 FR 67381-82). A number of the factors are already reflected in the proposed DBT rule text, such as requiring protection against suicidal attackers, insiders, and waterborne threats (70 FR 67382). Some of the factors are not included in the proposed text of the rule, such as the attribute of air-based threats (70 FR 67382). The Commission has received over one hundred comment letters, including comments on the consideration of the twelve factors. The NRC will address them as part of the final rulemaking determining the elements of the revised DBT specified in NRC regulations. This public rulemaking process fully comports with the requirements applicable to the Commission's conduct of rulemakings and is the mechanism by which the NRC will continue its consideration of all twelve factors.

12. Will you commit to informing the public how you considered these 12 factors, and whether you will revise the design basis threat to address them?

ANSWER:

Yes, the FRN for the Final DBT Rule will address the NRC's consideration and final action regarding each of the 12 factors included in the Energy Policy Act, as described in response to Question 11. That response will provide as much detail as possible to the public without compromising sensitive or classified information that has been integrated into the process of vetting each factor.

13. I continue to hear from constituents that changes to the hearing process have made requests more difficult and less likely to be granted. Since the changes, are you seeing a reduction in the number of hearing requests overall, and particularly in the number of successful requests that result in a hearing being granted?

ANSWER:

It is too early to tell how the number of hearings requested or granted will be affected. Although the most recent changes to NRC's hearing procedures, which became effective for proceedings noticed on or after February 13, 2004, did change certain procedural requirements, including the time frame in which petitions for leave to intervene and requests for hearing (including contentions) had to be submitted, the revisions did not include any substantive changes to the longstanding requirements for standing and the admissibility of contentions. The revised rule requires that contentions are now part of the initial petition for leave to intervene and request for hearing, but, at the same time, allows more time, 60 days after the publication of a notice of opportunity for hearing in the Federal Register, for submission of such petitions. Furthermore, because of the varying number and complexity of applications being considered during any given time period concerning facilities at different locations and with varied degrees of stakeholder interest, it is not possible to isolate meaningfully any specific factor that would generally result in a higher or lower total number of requests for hearing, or grants or denials of such requests. The revisions were implemented to enhance the efficiency and effectiveness of NRC adjudications while ensuring that the rights of all parties to fair, effective, and timely adjudications are maintained.

14. The Yucca Mountain repository is designed to house 70,000 metric tons of nuclear waste. By the year 2035, the U.S. is projected to produce 105,000 metric tons of nuclear waste from existing plants. Since the Nuclear Waste Policy Act requires the government to assume responsibility for permanently disposing of the nation's nuclear waste, we need to fully understand the impact of the current waste situation on the future of nuclear power generation?

NRC has said it wouldn't license reactors without reasonable confidence spent fuel can be safely disposed. Has NRC ever said success at Yucca was necessary for such confidence, and to keep licensing old and new reactors?

ANSWER:

No, it has not. In 1990, the Commission found reasonable assurance that at least one mined geologic repository would be available, somewhere within the U.S., within the first quarter of the 21st century. Later, in 1999, the Commission found no basis to reevaluate its earlier finding of confidence. The Commission decided that it would reevaluate its earlier Waste Confidence findings only when the impending repository development and regulatory activities had run their course or if significant and pertinent unexpected events occur, raising substantial doubt about the continuing validity of the1990 findings.

If DOE abandons the Yucca Mountain site, the Commission may need to reevaluate the 2025 availability date. Until such time, it would be inappropriate for the Commission to prejudge the outcome of a Yucca Mountain licensing proceeding or to speculate about the availability or acceptability of any alternative to Yucca Mountain.

15. In our full Committee hearing the week of March 1, 2006, we learned that DOE now does not have a firm deadline for submitting the Yucca Mountain application to the NRC. Is the NRC able to decide whether storage or disposal of high-level nuclear waste at Yucca Mountain will be safe without reviewing a full license application?

ANSWER:

Any Commission decision about the safety of storage or disposal at the proposed repository would be reached only after a comprehensive technical review of a license application and careful consideration of the record established in an adjudicatory proceeding.

16. The Administration is pursuing a new nuclear waste reprocessing program called the Global Nuclear Energy Partnership that could impact the amount and type of nuclear waste generated in the U.S. This new program relies on reprocessing technologies that are currently under development. Existing reprocessing technologies produce a byproduct which is a highly radioactive sludge-like residue that must be solidified and sealed in stainless steel canisters before it is shipped. Wouldn't this waste require special handling and wouldn't new regulations be required to govern its management?

ANSWER:

The NRC has the responsibility under Section 202 of the Energy Reorganization Act of 1974 to license facilities used primarily for the receipt and storage of high-level radioactive wastes resulting from activities licensed under the Atomic Energy Act or facilities authorized for the express purpose of long-term storage of radioactive waste generated by the Department of Energy which are not used as part of research and development activities. Although the NRC does not have regulatory authority over a DOE reprocessing facility, if a facility used to store the resulting high-level waste falls within NRC's jurisdiction, it is likely that NRC would find that some waste streams would require special handling and any necessary requirements to ensure the safe handling of the waste streams would be a part of the regulatory infrastructure developed by the NRC.

17. DOE is proposing to develop reprocessing technologies and build a reprocessing demonstration plant in the next 10 years. What is your position on whether the NRC would be responsible for licensing such a facility?

ANSWER:

Section 202 of the Energy Reorganization Act of 1974 defines NRC regulatory authority over DOE activities. Under the current law, the NRC does not have regulatory authority for, and would not license, any DOE reprocessing facility used to demonstrate the advanced recycling technology selected or any DOE facility used to reprocess commercial spent nuclear fuel. However, it should be noted that in 1974 all recent and contemplated reprocessing facilities for commercial spent fuel were under private sector control and subject to NRC licensing. In Section 202, Congress explicitly gave NRC authority over the Clinch River Breeder Reactor and other demonstration nuclear reactors, such as the burner reactor included in GNEP.

18. The first nuclear plant operating license will expire this year, approximately 10 percent will expire by the end of the year 2010 and more than 40 percent will expire by the year 2015. The Atomic Energy Act and NRC regulations limit commercial power reactor licenses to an initial 40 years but also permit such licenses to be renewed. Due to this selected period, however, some structures and components may have been engineered on the basis of an expected 40-year service life. How does the fact that some plants have an engineering design life of 40 years impact their ability to perform safely for potentially another 20 years?

ANSWER:

The Atomic Energy Act (AEA) permits the Nuclear Regulatory Commission (NRC) to issue operating licenses with terms up to 40 years. The AEA limits the duration of operating licenses for nuclear power plants to a maximum of 40 years, but permits renewal of the licenses. The original 40-year license term was selected on the basis of economic and antitrust considerations, not design or operational limitations. However, once established, the designs of some structures, systems, and components within the plant were subsequently based on a 40-year operating life.

The license renewal rule (10 C.F.R., Part 54), focuses the NRC's license renewal review on the effects of aging on the functionality of certain plant systems, structures, and components in the period of extended operation and a few other issues related to safety during extended operation. The NRC believes that sufficient technical understanding of age-related degradation exists to enable licensees to develop activities for ensuring safe operation of their plants for the additional 20 years beyond expiration of their existing licenses.

If a licensee chooses to apply for license renewal, the application must provide the NRC an assessment of the technical aspects of plant aging and must describe how the aging will be managed. Time-limited aging analyses within the scope of the rule that specifically rely on the assumption of a 40-year operating life must also be re-evaluated. In addition, the licensee must also prepare an evaluation of the potential impact on the environment to support plant operation for an additional 20 years. The NRC documents its reviews in publicly available documents and performs verification inspections at the licensee's facilities before making a decision on issuing a renewed license. Therefore, the NRC's license renewal process provides reasonable assurance that aging will be managed for all structures, systems, and components within the scope of the rule (including those with an initial specified design life of 40 years) such that they will continue to perform their required safety functions for the period of extended operation.

QUESTIONS FROM SENATOR ISAKSON:

1. During the hearing, I brought up the issue of potassium iodide, but didn't get a chance to pursue my question with the Commission. It is my understanding that the Department of Health and Human Services (HHS) has made a recommendation to expand the stockpiling of potassium iodide beyond the 10-mile radius around a nuclear facility which is the current requirement. Please provide the Commission's position on the HHS's recommendation for the record.

ANSWER:

Based on the NRC's decades of experience with nuclear power plant emergency preparedness and radiological protection of the public, it is the NRC's conclusion that expanded distribution of potassium iodide (KI) is unnecessary. Expanded distribution of KI is unnecessary because of the current, well-established, and scientifically sound framework of the NRC's emergency preparedness regulations. This framework includes predetermined protective actions for populations within the 10- and 50-mile ingestion exposure pathway Emergency Planning Zones (EPZs) to provide the necessary protection of public health and safety. These predetermined protective actions include interdiction of contaminated milk, food, and water, as well as protective measures for livestock. NRC's conclusion is supported by a January 2004 study by the National Academy of Sciences, which found that food testing and interdiction programs in place throughout the United States are more effective preventive strategies than expanded distribution of KI for ingestion pathways. Additionally, many States and other interested entities, including Federal agencies, have expressed opposition to the distribution of KI beyond the existing 10-mile EPZs.

Additional detail on the Commission's position on HHS's draft guidelines for expanded KI distribution are provided in the November 1, 2005 letter from Mr. William Kane, NRC's Deputy Executive Director for Reactor and Preparedness Programs, to Dr. Claypool of HHS's Office of Mass Casualty Planning, which is attached (Attachment 7) for your convenience.

QUESTIONS FROM SENATOR OBAMA:

1. Will you please provide me with a list of the other tritium leak incidents elsewhere in the country over the past 10 years, including location, and level of radiation?

ANSWER:

This information is being compiled as part of several issues being addressed by the task force the NRC created to examine the issue of inadvertent, unmonitored releases of radioactive liquid containing tritium from nuclear power plants. The NRC has deemed it necessary to do a broad review to determine the extent of the issue and to recommend possible agency actions. Specifically, the Task Force will conduct a review of known inadvertent releases (1996 to present) of radioactive liquid to the environment at power reactor sites, including power reactors in decommissioning. At this time, Attachment 8 presents a preliminary listing of events that have included tritium leaks.

The updated information will be included in a report expected to be released in September 2006. The creation of the Task Force and its responsibilities are posted as a news release item on the NRC website (www.nrc.gov). The news release is attached (Attachment 9).

2. Will you please provide me with the NRC's views on the Nuclear Release Notice Act (S. 2348), which I introduced earlier this month?

ANSWER:

The Nuclear Regulatory Commission supports the notification objectives of S. 2348 but believes certain changes in S. 2348 are desirable.

Section 2 of the bill would amend the Atomic Energy Act to require that in the case of certain unplanned releases, licensees of utilization facilities "shall immediately notify the Commission, and the State and county in which the facility is located, of the release." The required "immediate" notification to States and counties does not appear commensurate with other NRC notification requirements. For example, 10 C.F.R. 50.72 (a)(3) requires that after a licensee declares an emergency within one of the Emergency Classes, the licensee must notify the affected State and county and the NRC no later than one hour after the time of the declaration. Unplanned releases below the level of an emergency present a substantially smaller risk to the public. Unplanned releases that are a microscope fraction of the facility's normal releases pose a minimal risk to the public. Furthermore, the nature of unplanned releases is such that it may take substantial time to determine whether such a release has occurred and its potential impacts.

Section 2 of the bill would also apply the reporting requirements to any unplanned release of quantities of fission products or other radioactive substances within allowable limits for normal operation established by the Commission or other applicable Federal laws or standards but that occurs more than twice within a 2-year period originating from the same source, process, or equipment at a facility. The breadth of the definition of "unplanned release" raises scope issues. Additionally, the proposed legislation does not set a lower bound for recurrent unplanned releases within allowable limits that are subject to the reporting requirements. For example, an unplanned pathway release of several thousand gallons of contaminated liquid could be readily recognized and tracked for possible reporting. If a second event occurs for the same system but only results in the release of one gallon of contaminated liquid, it seems questionable whether, as the bill requires, such a release should trigger reporting requirements. Also, the wording "originating from the same source, process, or equipment at a facility" lacks precision. This could cause confusion for tracking "releases" and determining whether the criterion of two occurrences within a 2-year period has been met.

3. Mr. Diaz, you stated that the NRC is taking a comprehensive look at the tritium problem, including the way it is monitored, the environmental situation, and communications between the NRC and state environmental protection agencies. Do you expect this analysis to be completed by Memorial Day? Upon completion of this analysis, will you please submit it to the members of this Committee?

ANSWER:

An NRC Task Force has been established to examine the issue of inadvertent releases of radioactive liquid to the environment at power reactor sites, highlighted by recently identified incidents at Braidwood, Indian Point, Byron, and Dresden. Given the mandate of the Task Force, the report is expected to be released in September 2006. When the report is issued, it will be provided to you and the other members of the Committee.

4. On March 13, 2006, almost days after our hearing, approximately 200 gallons of water spilled at the Exelon Braidwood station from an on-site tank where radioactive liquids are temporarily being stored in the wake of the recent issue on tritiated water leaks. I am told that testing onsite of the water in the berm area showed about 255,000 picocuries per liter. I understand that the leakage was not reportable to the NRC, but the licensee has informed State and local officials and issued a news release.

    a) Why was this leakage not reportable to the NRC? What thresholds for reporting were not met that otherwise would require NRC reporting?

    b) Are there reporting thresholds that differentiate between releases that occur on the licensee property as opposed to off-site?

ANSWER:

As indicated in the question, the spill which occurred on March 13, 2006, at the Braidwood site involved approximately 200 gallons of water with low levels of tritium contamination. The total amount of tritium in the water was about a millionth of that found in a typical exit sign. Some of the water, remaining on the surface, was collected and pumped back into the reinforced berm, and ultimately into a storage tank. The water spilled from the berm remained in the immediate vicinity, and there was no evidence of an off-site release of the slightly contaminated water.

NRC's reporting requirements include thresholds that are linked to potential impacts on radiological protection of public health and safety. The types of events discussed in the question, involving low concentration, localized spills with no off-site impacts, are not required to be formally reported to the NRC because the thresholds of the reporting requirements contained in NRC regulations were not met. These requirements are more specifically described below and focus on emergency events and events involving exposures to individuals in excess of regulatory requirements or significant releases of radioactive materials in excess of regulatory requirements, both on site and off site.

Although not required by NRC regulations, the on-site resident inspector was informed of the event, and NRC staff from the Regional office, in addition to the resident inspector, conducted follow-up activities. To put this communication in context, the resident inspectors located at the plant maintain a day-to-day awareness of plant activities, including routine and non-routine evolutions and occurrences. As such, there is on-going dialogue relative to issues at the plant as well as issues which may be of particular NRC or public interest.

In addition, NRC's regulations in 10 C.F.R. 20.1501 require the licensee to perform radiological surveys and evaluations that are necessary to ensure compliance with NRC requirements and to evaluate the magnitude and extent of radiation levels, the concentrations or quantities of radioactive material, and the potential radiological hazards. The licensee is expected to evaluate on-site spills, and these evaluations are subject to NRC review and inspection.

NRC regulations include several requirements addressing NRC notification of incidents and accidents. As indicated above, these requirements primarily focus on actual or potential doses to individuals and significant releases. These requirements are as follows:

i. 10 C.F.R. 20.2202 (Notification of incidents). Examples of the types of notifications required under this part include immediate notification of exposures to individuals (25 rem) and releases of radioactive material, inside or outside the restricted area, so that, if an individual was present for 24 hours, the individual could inhale radioactive materials in excess of five times the annual limit.

ii. 10 C.F.R. 20.2203 (Reports of exposures, radiation levels, and concentrations of radioactive material exceeding the constraints or limits). Examples of the types of notifications required under this part include reporting within 30 days of an individual member of the public receiving a radiation dose of 100 mrem or more, and concentrations of radioactive material in restricted areas in excess of 10 times the applicable limit.

iii. 10 C.F.R. 50. 72(b)(2)(xi) (Immediate notification requirements for operating nuclear power reactors) requires the licensee to notify NRC within 4 hours of any event related to the health and safety of the public, or protection of the environment, for which a news release is planned or notifications have or will be made to other government agencies. As noted earlier, the Braidwood event was communicated to our resident inspector on site and follow-up was conducted by the NRC staff on site and from the Regional office.

iv. 10 C.F.R. 50.73 (Licensee event reporting system). An example of the type of notifications required under this part includes a 30-day written report for any liquid effluent release that, when averaged over a time period of 1 hour, exceeds 20 times the applicable concentration for all radionuclides except tritium and dissolved noble gases.

For nuclear power plants, the operating license also requires that, as part of the Radiological Environmental Monitoring Program (REMP), the licensee implement a program for the control and monitoring of liquid and gaseous radiological effluents released to off-site locations. The REMP identifies reporting levels for radioactivity identified in off-site environmental samples, including levels for tritium as one among other radionuclides in water. As an example for tritium, the reporting levels are 20,000 picocuries per liter for drinking water and 30,000 picocuries per liter for non-drinking water. The REMP is one of several requirements mandated by the operating license in each plant's radiological effluent technical specifications.

Although not specifically a reporting requirement, the NRC also has record keeping requirements associated with on-site radiological conditions. These requirements are contained in 10 C.F.R. 50.75(g) and relate to site decommissioning. For this, the licensee is required to maintain records of spills or other unusual occurrences involving the spread of contamination in and around the facility or site. These records are maintained for the life of the plant and are subject to NRC inspections.

Attachments:
Attachment 1 - Letter to Senator George Voinovich on the NRC's current space requirements
Attachment 2 - Industry Trend Results
Attachment 3 - Operating Experience: Unplanned Reactor Shutdowns (6 months or longer)
Attachment 4 - Press Release on the Maine Yankee Independent Safety Assessment Results
Attachment 5 - Attachment A to the Vermont Yankee report
Attachment 6 - Letter to Senator Clinton on an Independent Safety Assessment at Indian Point
Attachment 7 - Letter to Dr. Robert Claypool on Commission's position on HHS's draft guidelines for expanded KI distribution
Attachment 8 - Preliminary listing of Events Involving Tritium Leaks
Attachment 9 - Press Release on the Creation of a Liquid Radioactive Release Lessons-Learned Task Force

April 5, 2006

The Honorable George V. Voinovich
Chairman, Subcommittee on Clean Air,
Climate Change and Nuclear Safety
Committee on Environment and Public Works
United States Senate
Washington, D.C. 20510

Dear Mr. Chairman:

Thank you for the opportunity to testify before the Committee on March 9, 2006, on the U.S. Nuclear Regulatory Commission's (NRC's) programs. During the hearing, Senator Carper asked how the Committee can help the NRC better accomplish its mission. In response, the Commission agreed to provide the Committee specific information on current space requirements and on legislation needed in this area that would support our accelerated hiring program for the work associated with new reactor licensing. The Committee's assistance in two specific areas would be of great value to the NRC: legislative authority for the General Services Administration (GSA) to acquire immediately space as close as possible to the NRC headquarters location and legislation relief to accelerate the space acquisition process.

The NRC will soon exhaust all available office space in the current headquarters buildings despite concerted efforts to utilize all available space for workstations. The space shortage is particularly acute because the NRC needs to expand its staff to accommodate anticipated new work. The Commission recommends legislative action that would provide GSA the authority to acquire immediately NRC headquarters space as close as possible to our current location.

Congressional action is also needed to release GSA from the competition requirements of the acquisition process. GSA has been responsive to our February 10, 2006, request detailing our requirements for an additional 100,000 square feet of permanent building space. We are working with them to prepare the necessary documentation to acquire the space through the established portfolio acquisition process for Congressional approval. However, this process will not result in occupancy of our permanent building until FY 2009 at the earliest, which could deprive the NRC of a unique opportunity to extend our headquarters campus to the property very close to both buildings that is currently under development. The recommended language to accomplish both purposes is enclosed.

The property under development would provide sufficient office space to meet NRC's permanent needs. Acquiring this space would allow the NRC to maintain the consolidation benefits achieved in 1994 when the dispersed headquarters staff were finally consolidated in Rockville. One of the key benefits involves maintaining our incident response capability, which requires immediate assembly of technical staff from various NRC offices. It also provides a unique opportunity for NRC, GSA, and the developer to align the design and construction plans and lease terms as necessary to accommodate NRC's evolving space needs in an efficient and cost effective manner. To acquire this space, GSA must engage the developer in negotiations now, well ahead of the portfolio acquisition process and subsequent competitive acquisition schedule, or run the risk of losing the space to another interested party.

For many years, NRC was located in as many as eleven different locations in Montgomery County and Washington, D.C. The impact of these many detached locations was highlighted in NRC's Three Mile Island lessons learned report and in General Accounting Office (now Government Accountability Office) reports that cited the need for consolidation as essential to NRC's regulatory effectiveness and efficiency of operations. After many years of effort and with the support of Congress and the GSA, the NRC achieved consolidation at its two headquarters buildings in 1994.

Keeping the NRC staff consolidated substantially enhances NRC's ability to discharge our regulatory responsibilities. The Commission urges you to consider a legislative solution that would enable GSA to accelerate the space acquisition process.

The Commission appreciates the help and support that this Committee has provided, and we look forward to continuing to work with you in the future.

Sincerely,

/RA/

Nils J. Diaz

Enclosure:
As stated

cc: Senator Thomas R. Carper

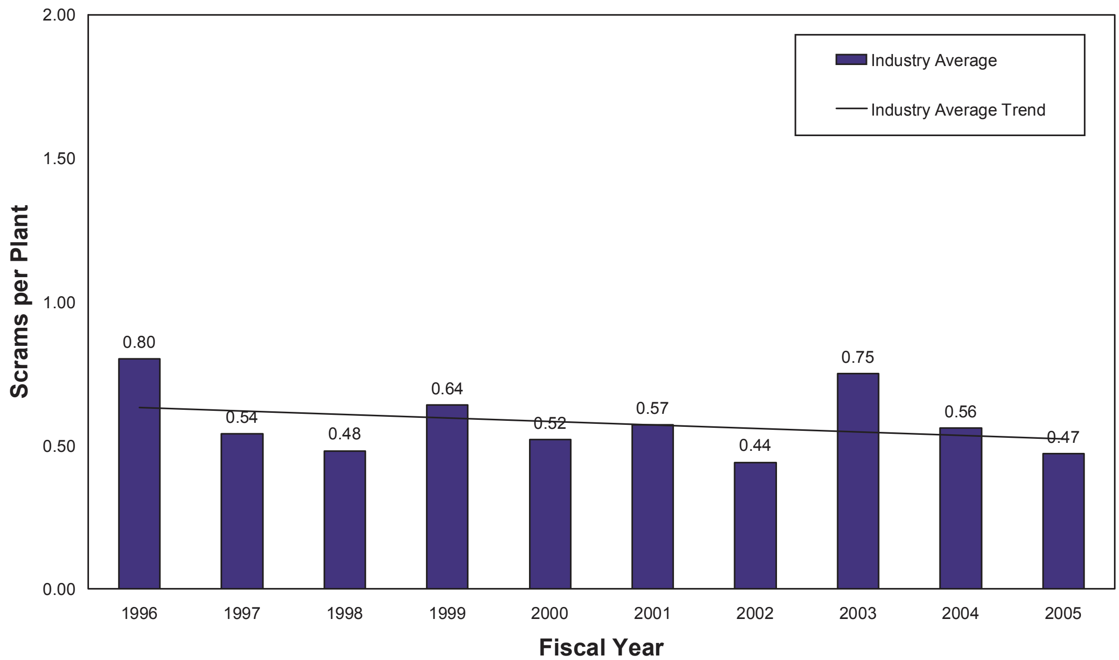
Recommended Legislative Language

Notwithstanding any other provision of law relating to the acquisition or lease of real property, the General Services Administration is authorized to arrange for additional office space for the U.S. Nuclear Regulatory Commission (NRC) headquarters employees. Such space shall be as close as reasonably possible to the existing NRC campus in Rockville, Maryland, as determined by the NRC, to maintain NRC's regulatory effectiveness, efficiency, and emergency response capability.

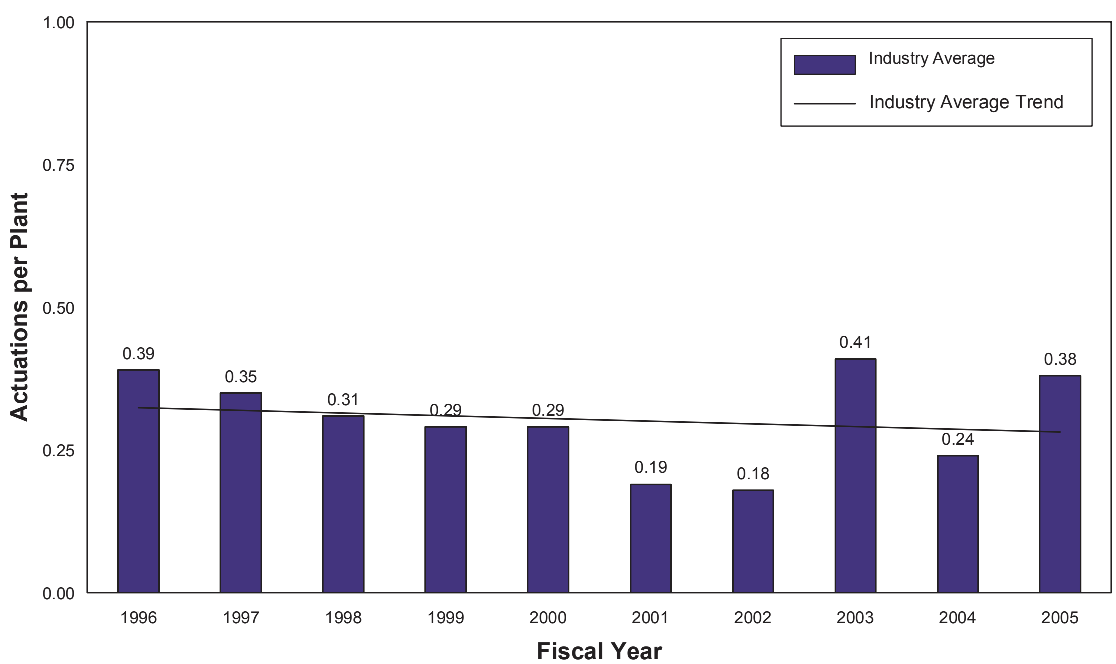
Enclosure

FY2005 Long-Term Industry Trend Results

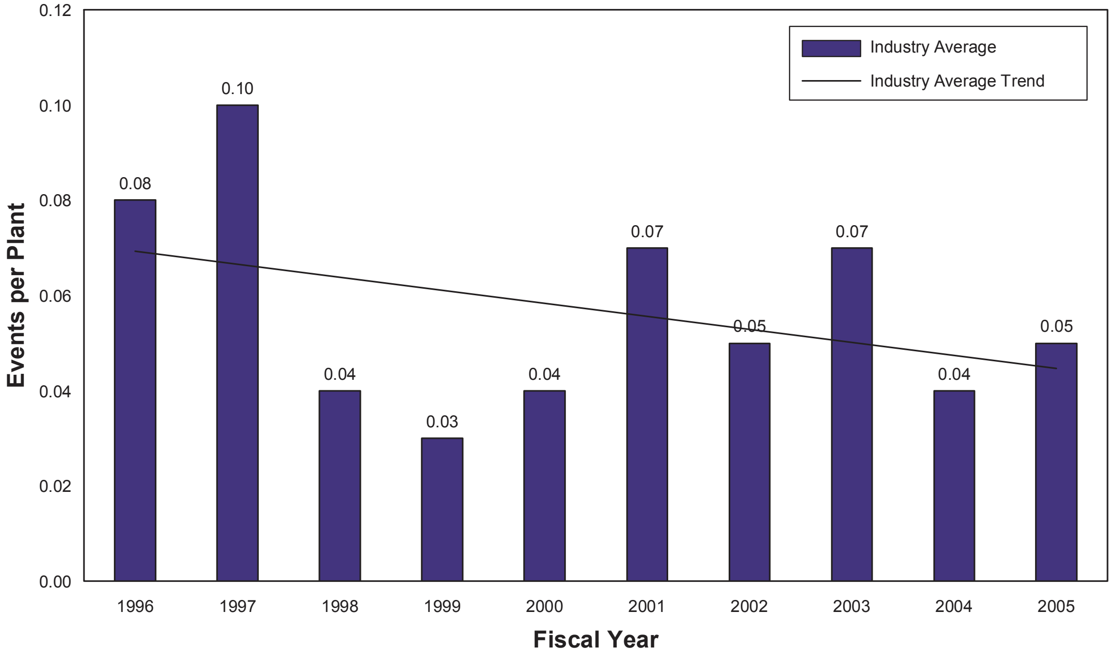
Automatic Scrams While Critical



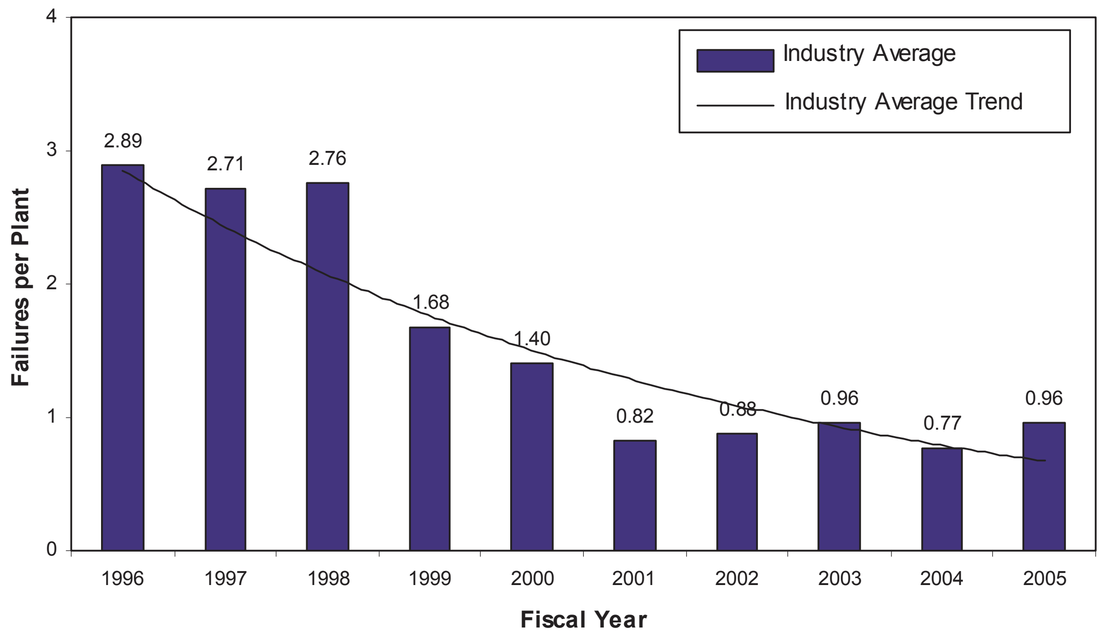
Safety System Actuations



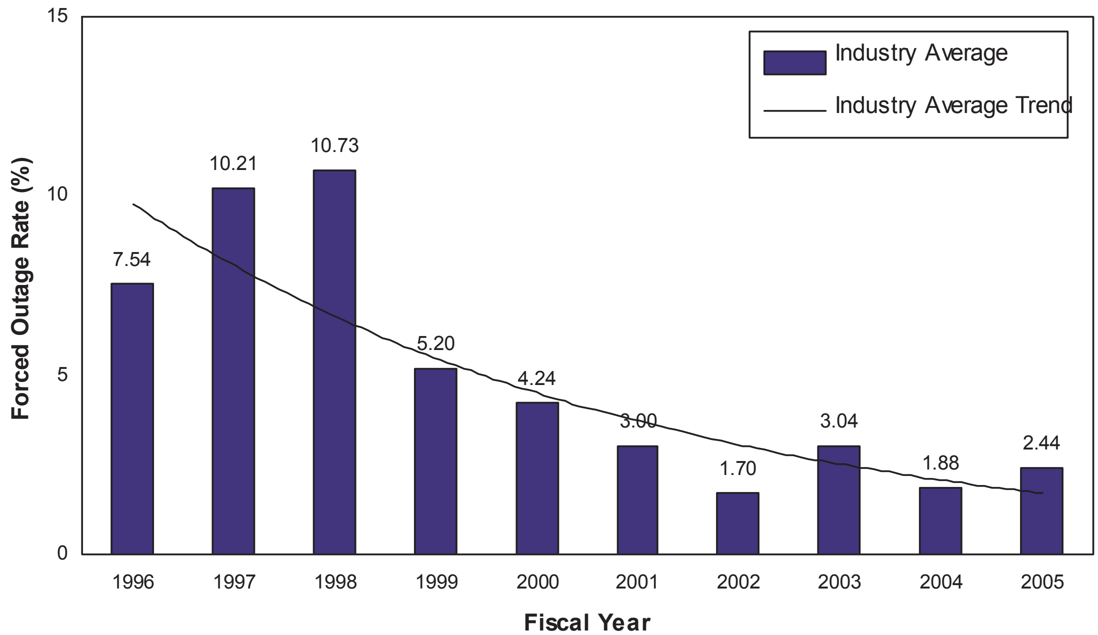
Significant Events



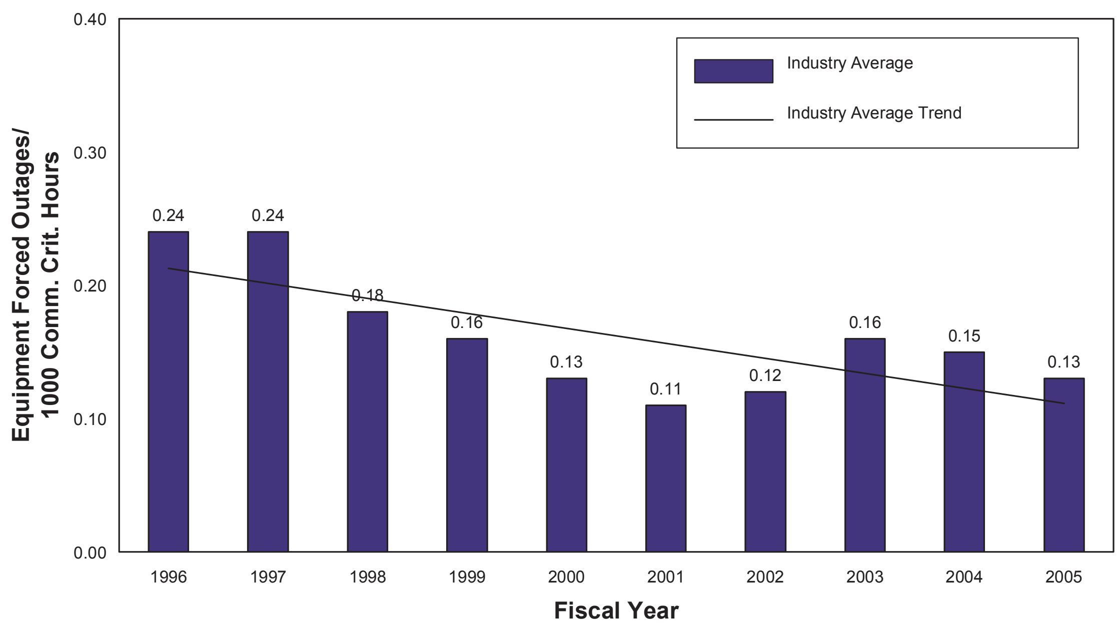
Safety System Failures



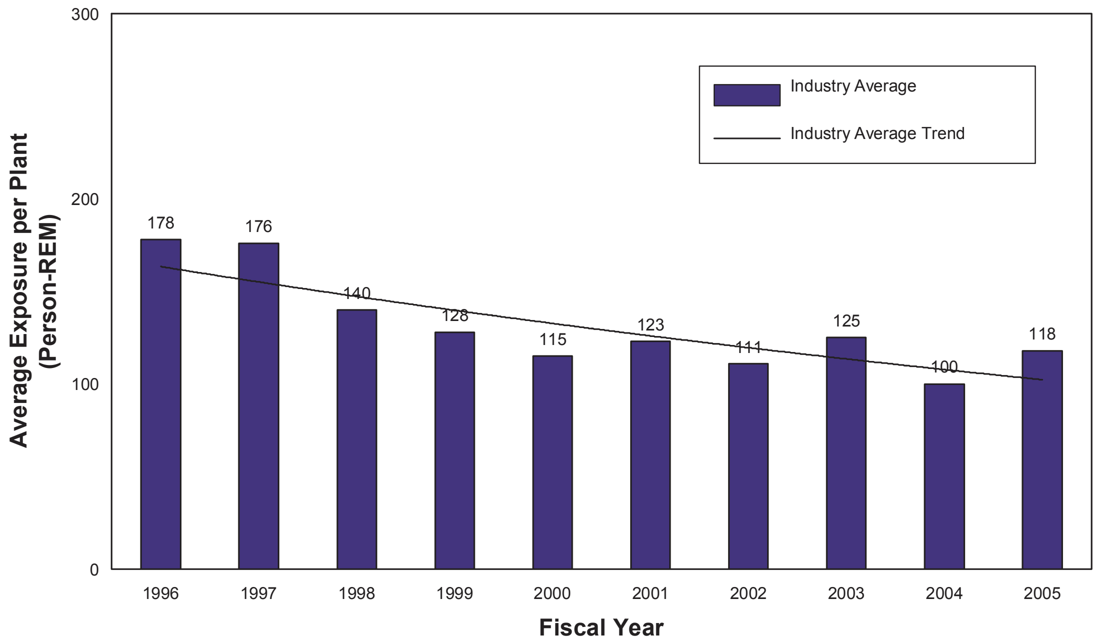
Forced Outage Rate (%)



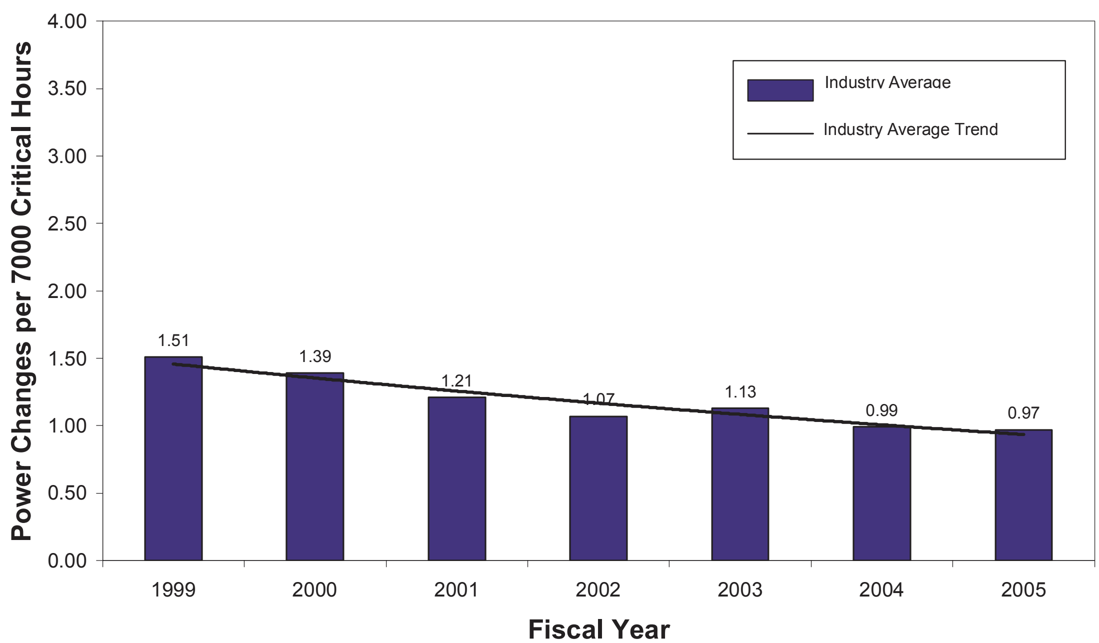
Equipment Forced Outages/ 1000 Commercial Critical Hours



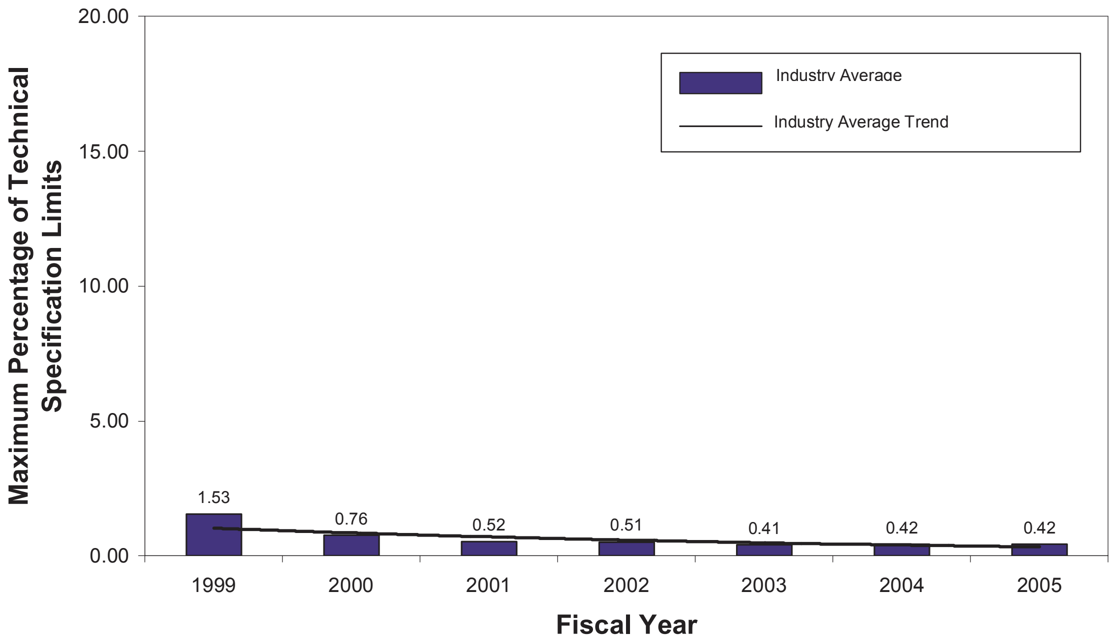
Collective Radiation Exposure



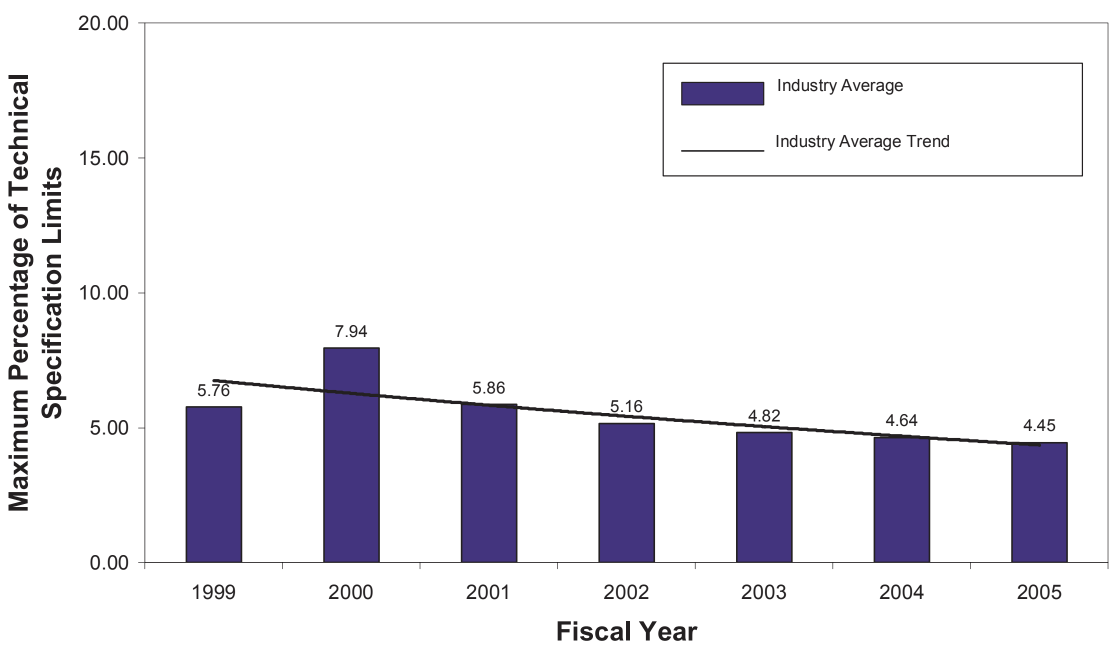
Unplanned Power Changes



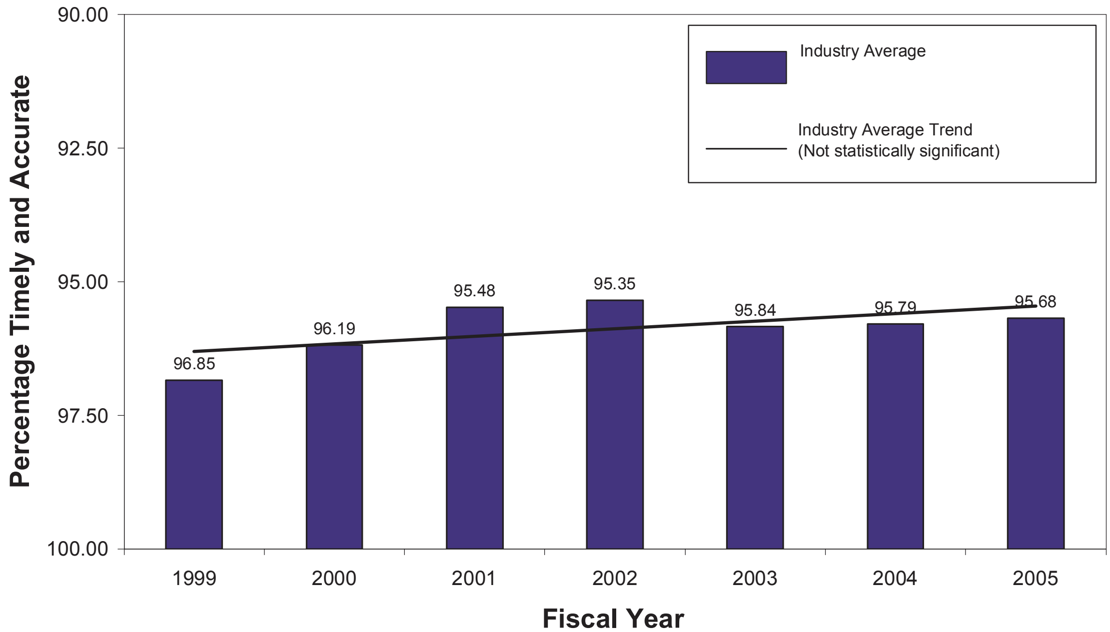
Reactor Coolant System Activity



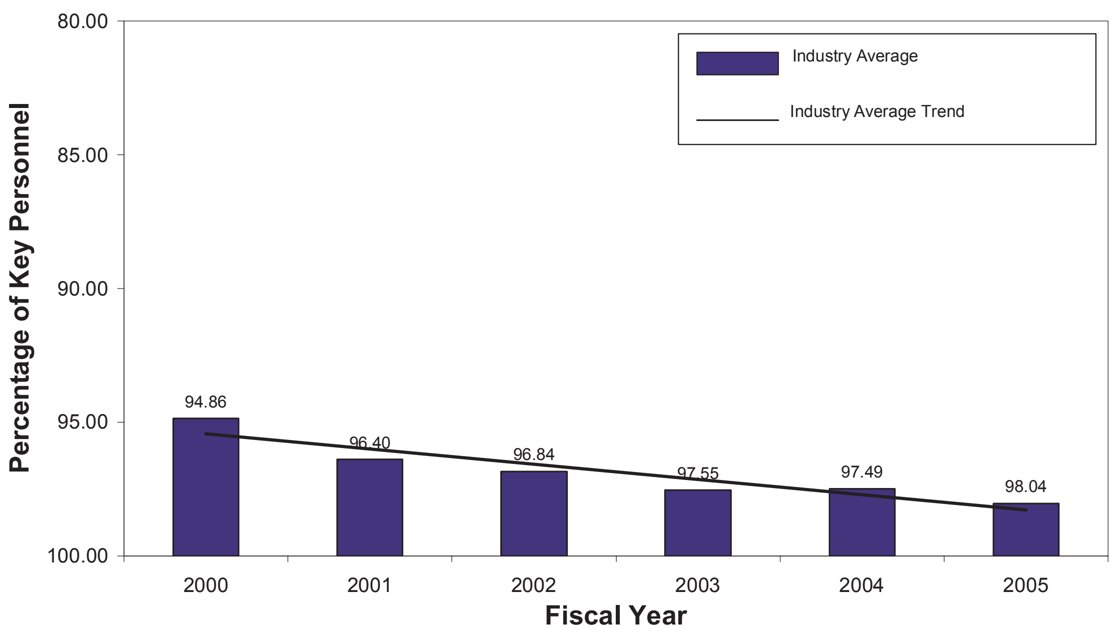
Reactor Coolant System Leakage



Drill/Exercise Performance

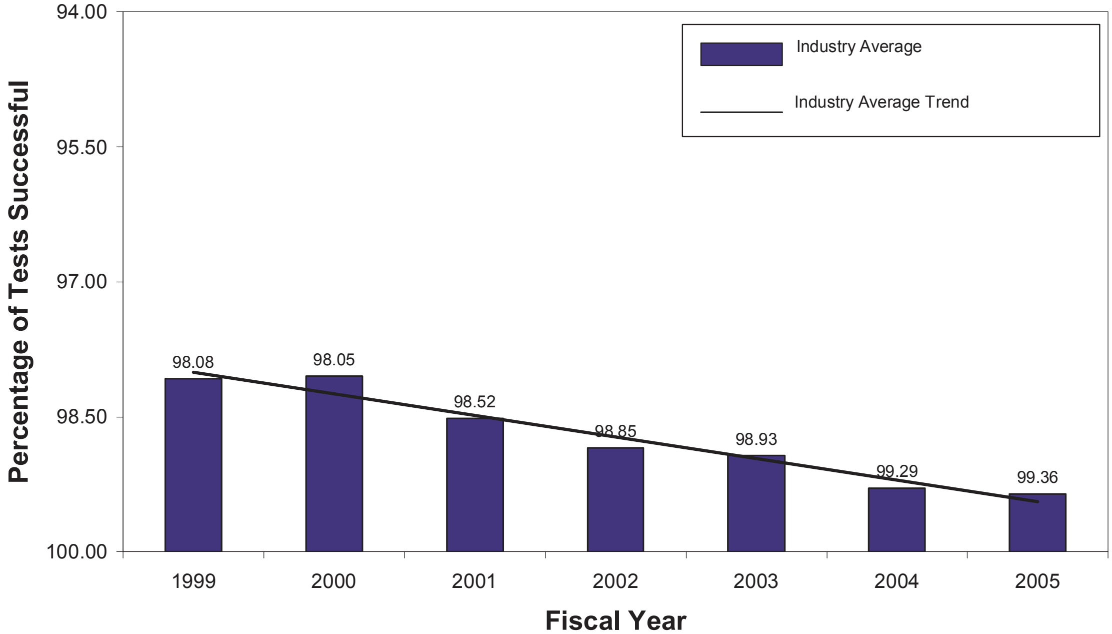


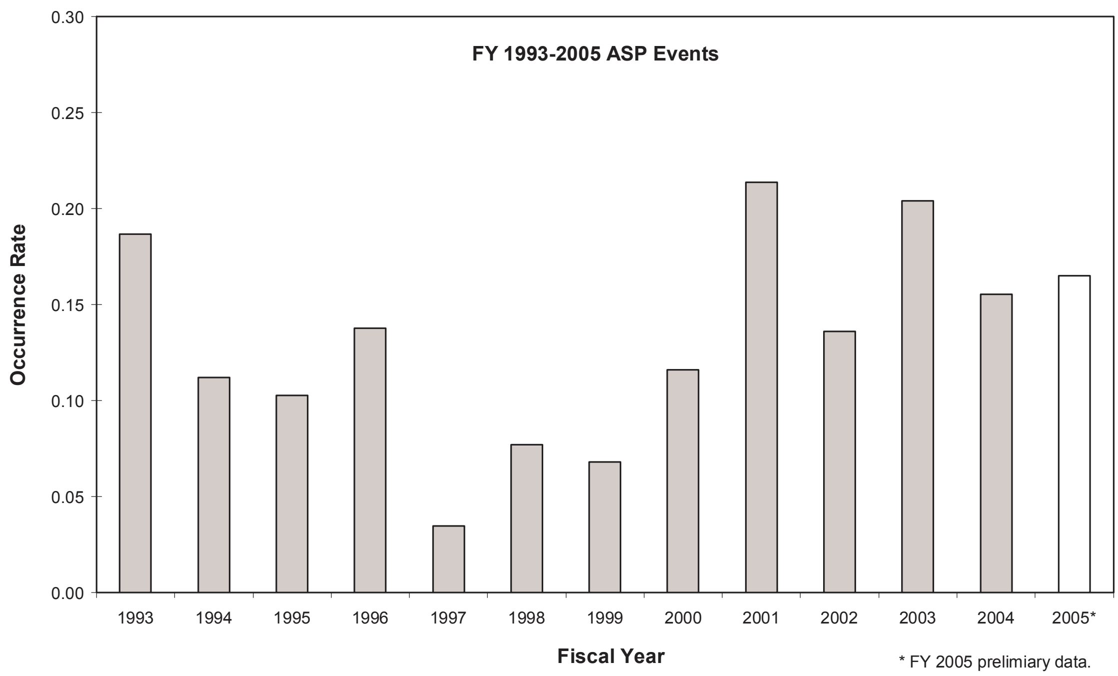
ERO Drill Participation



Note: Complete Fiscal Year 1999 Data not Available

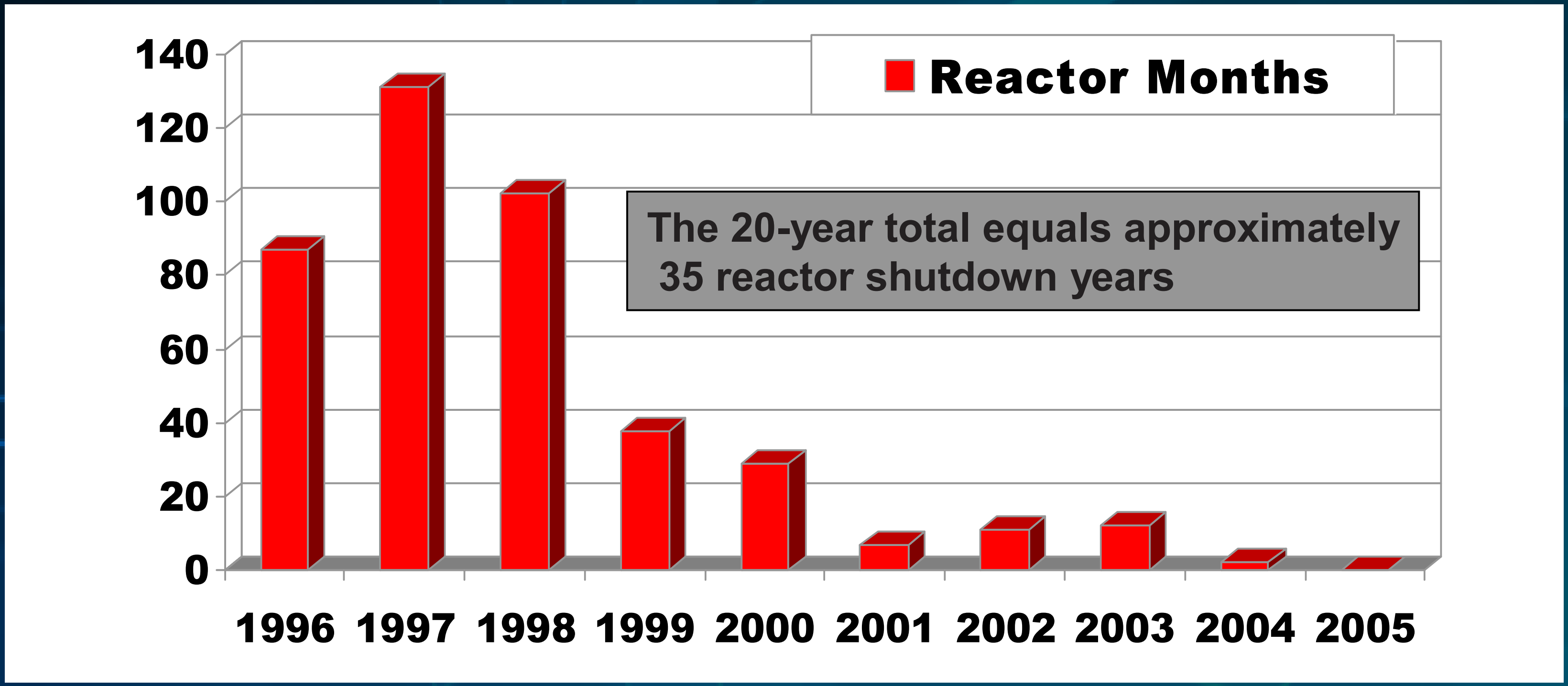
Alert and Notification System Reliability





Total Precursors - occurrence rate by fiscal year. No trend line is shown because no trend was detected that was statistically significant.

Operating Experience: Unplanned Reactor Shutdowns (6 months or longer)



United States Nuclear Regulatory Commission
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Washington, DC 20555
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No. 96-133 FOR IMMEDIATE RELEASE
(Tuesday, October 8, 1996)

INDEPENDENT SAFETY ASSESSMENT OF MAINE YANKEE RATES OPERATIONS ADEQUATE WITH SIGNIFICANT WEAKNESSES AND DEFICIENCIES

A Nuclear Regulatory Commission Independent Safety Assessment team has concluded that operations at the Maine Yankee nuclear power plant are adequate, but identified a number of significant weaknesses and deficiencies that will result in violations. The plant, operated by the Maine Yankee Atomic Power Co., is about 10 miles north of Bath, Maine.

"These weaknesses and deficiencies appear to be related to two root causes: economic pressures to contain costs and poor problem identification as a result of complacency and a lack of a questioning attitude," NRC Chairman Shirley Ann Jackson said in a letter written to the licensee, which accompanies the report.

In its report, the team says adequate safety margins exist to operate the plant at its originally licensed power level of 2440 megawatts thermal (Mwt). But the team did not conclude, and the licensee did not demonstrate, that the plant can be safely operated at its more recently licensed output of 2700 Mwt. Since January, the plant has been restricted to 2440 Mwt by an NRC confirmatory order.

According to the report, economic pressures limited resources and interfered with the utility's ability to complete projects and other efforts that would improve plant safety and testing activities. Examples included a failure to adequately test safety-related components; long-standing deficient design conditions, such as an undersized atmospheric steam dump valve; issues involving environmental qualification of electrical equipment; and the lack of effective improvement programs.

The team said complacency and the failure to identify or promptly correct significant problems were apparent as demonstrated by previously undiscovered deficient conditions of the service water and auxiliary feedwater systems. Other weaknesses cited included inadequacies in ventilation systems, post-trip reviews that lacked rigor and completeness, and emergency operating procedures that may not have adequately addressed an inadequate core cooling event and a steam generator tube rupture under certain conditions. The report said Maine Yankee Atomic Power Co. lacked a questioning attitude during test performance and evaluation that was not conducive to discovering equipment problems, but rather to accepting equipment performance. Self-assessments, by the utility, the team concluded, occasionally failed to identify weaknesses, or incorrectly characterized the significance of findings.

The 25-member team, headed by Edward L. Jordan, the NRC's Director of the Office of Analysis and Evaluation of Operational Data, began its work in June after concerns were raised by the NRC's Inspector General about the adequacy of the licensee's use of the Relap/5YA computer code and NRC inspection activities at the site. The team was formed at the request of Chairman Jackson. The inspection also was responsive to concerns expressed by Gov. Angus King of Maine. Members included NRC specialists, expert contract support from a national laboratory, independent consultants and three representatives from the State of Maine.

Other findings include the following:

-- Maine Yankee was in general conformance with its licensing basis, although significant items of non-conformance were identified. The plant's licensing basis, although understood by the utility, lacked specificity, contained inconsistencies and had not been well maintained. However, the quality and availability of design-basis information was good overall, the team concluded.

-- Performance in the area of operations was very good, with strengths noted in operator performance during routine and abnormal operating conditions. But some compensatory measures unnecessarily burdened operators or complicated their response to abnormal conditions. Additionally, log-keeping practices were not rigorous.

-- Maintenance was rated as good overall, but testing was weak. The results of the review of equipment reliability for the auxiliary feedwater, emergency feedwater, high pressure safety injection and emergency diesel generator systems showed mixed equipment performance. Although the plant's material condition was good overall, a number of significant material condition deficiencies were noted, along with a decline in material condition following a 1995 shutdown to repair steam generator tubing.

-- The quality of engineering work was mixed but considered good overall. Strengths were noted in the capability and experience of the engineering staff, day-to-day engineering support of maintenance and operations and in the quality of most calculations. However, engineering was stressed by a shortage of resources, and there was a tendency to accept existing conditions.

A copy of the executive summary of the inspection report is available upon request from the NRC's Office of Public Affairs. The full inspection report has been posted on the Internet at this address: http://www.nrc.gov/OPA/reports.

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ATTACHMENT A

Summary of Items Reviewed

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
115 kV - Breaker K1	Transformer T-4 feed to 115 kV bus: required to supply power from the 345 kV switchyard to the Startup Transformers.	No automatic actions required except fault clearing; safety busses would disconnect or be prevented from connecting to circuit after a fault.
115 kV - K.1 Logic Relay	RCIC logic relay K.1 fails to operate on demand. Rationale: Malfunction of RCIC turbine trip instrumentation could cause loss of RCIC System.	The inspectors found no specific operator action for this component and that a failure of the logic relay would result in control room alarms which would be responded to by the operators. The inspectors found that related control room alarms were functioning properly, and that the associated alarm response procedures were current.
125 V Battery B-1 and A-1	Station Battery: Supplies power to the station 125 VDC loads when the battery chargers are not available.	Detailed review completed.
24 Vdc - ES-24DC-2	Power Supply Converter: Supplies power to the 24 VDC ECCS Analog Trip System.	No low margin or other issues identified.
345 kV - Breaker 381-1	Northfield 345 kV line to 345 kV North Bus: required to provide power from the Northfield 381 to the 345 kV switchyard.	Detailed review completed.
4 Kv - Breaker 12	Bus 1 Feed Breaker from UAT: required to open on generator trip to enable access of one safety train to the offsite source through the SUT	No low margin issues identified.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
4 Kv - Breaker 13	Bus 1 Feed Breaker from SUT: required to close on generator trip to enable access of one safety train to the offsite source through the SUT .	Detailed review completed.
4 Kv - Breaker 22	Bus 2 Feed Breaker from UAT: required to open on generator trip to enable access of one safety train to the offsite source through the SUT.	The inspectors found that the only operator action for this component was breaker open/close operation. Additionally, the inspectors found that the related control room alarms were functioning properly and that the associated alarm response procedures were current. The inspectors found no issues with this component related to operator actions.
4 Kv - Breaker 23	Bus 2 Feed Breaker from SUT: required to close on generator trip to enable access of one safety train to the offsite source through the SUT.	Detailed review completed.
4 Kv - Breaker 3V	Vernon Supply Breaker to Bus 3: required to supply power from the Alternate AC Power source to one 4160V safety bus.	No specific issues identified with breaker. Other issues reviewed as part of overall Station Blackout Capability.
4 Kv - Breaker 3V4	Vernon Tie Breaker: required to supply power from the Alternate AC Power source to either 4160V safety bus.	Detailed review completed.
4 kV UV Relays	4160V Undervoltage Relays: required to provide adequate voltage to safety-related AC loads, reset setpoint must be optimized to prevent spurious loss of offsite power.	Detailed review completed.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
69 kV - Vernon Generator	Vernon Hydroelectric generator station: required to supply power from the Alternate AC Power source to either 4160V safety bus.	Detailed review completed.
69 kV to 4160 V Vernon Transformer	Vernon Tie Transformer: required to supply power from the Alternate AC Power source to either 4160V safety bus.	Detailed review completed.
125 VDC Distribution Panels	Supplies 125 VDC loads.	Detailed review completed.
Alignment of RHRSW to the RPV	Operator fails to align the RHRSW injection to RPV.	Aligning RHRSW injection to the RPV is one of the methods which can be used for RPV injection to prevent core damage in accordance with EOPs given an ATWS scenario. The validated time through simulator observation was 1 minute to complete the actions for alignment. Additionally, prior to using RHR SW for RPV injection, other systems such as condensate/feedwater , CRD, and RHR will be used to attempt to fill the RPV. The operators are regularly trained and evaluated in this event scenario further reducing the likelihood of the task not being completed within the required time.
Bus Transfer Scheme	Circuit breakers, synchronism check relays, timing relays, and voltage relays required to enable transfer of 4160V buses from the Unit Aux Transformer to the Startup Transformers.	Detailed review completed.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
Closure of Vernon Tie Breakers	Operator fails to close the Vernon tie breakers.	One of the primary AC power recovery actions in the event of a loss of normal power is to use the dedicated tie line from the Vernon hydro Station to power either 4260VAC Bus 3 or 4 (vital power). The action is performed by the operators in the main control room by manipulating switches for 2 DC powered breakers. Validation studies and operator observation in the simulator have shown that the task can be accomplished in less than 4 minutes. Adequate margin exists currently and for the CPPU to accomplish the action. Additionally, operator response to loss of power events is trained regularly in the simulator and classroom. While no issues identified with VY operator actions, a finding was identified with the licensee's overall station blackout response.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
Condensate Pump	Review condensate operation before and after the power uprate (including recirc pump runback modification). The Condensate and Feedwater system does not directly perform any safety-related function. Portions of the Feedwater system and check valves provide Reactor Coolant Pressure Boundary and Containment Isolation functions. The condensate pumps 1) supply water to the Feedwater pumps and 2) provide sufficient NPSH for operation of the FW pumps. The loss of a condensate pump could be a contributing factor to a transient initiation. The condensate pumps are directly impacted by the EPU due to the need to increase the flow volume by approximately 20%.	No low margin or other issues identified.
Containment Pressure	During a loss of coolant event or an ATWS the containment pressure will be elevated and the suppression pool level will increase.	Detailed review completed.
CST Transient Analysis Temperature Non-conservative	Transient analysis Condensate Storage Tank Temperature non-conservative compared to actual maximum operating temperatures. This issue stems from a similar event at Point Beach.	Detailed review completed.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
CST Level Instrumentation	Rationale: Important for maintaining required CST inventory for RCICS and controlling automatic transfer of RCICS suction to the suppression pool.	Detailed review completed.
CV-109	Failure of check valve CV-109 (valve between the N2 bottle and the SRV) to open. Failure of this check valve to open will prevent N2 supply to the Main Steam Safety Relief Valves.	Detailed review completed.
CV-19	RCIC check valve CV-19 (RCIC suction check valve from the CST) fails to open on demand. This valve must open to provide flow from CST to RCIC pump suction, and close to prevent flow from torus to CST during RCIC pump suction transfer.	A detailed review was not performed for this check valve because no performance problems were indicated from the maintenance history.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
CV-2-1A, 1B, 1C	RFP discharge check valves. They are risk significant because if they fail to close following an RFP trip they could make other RFPs inoperable. Prior to EPU two pumps are operational. After EPU three pumps will be operational. When two pumps are operational, one of the MOVs, 4A, 4B or 4C will be closed for the non-operational pump as such, this is not a current potential event. However, after EPU the third valve will not be closed thus this is a potential failure scenario.	A detailed review was not performed for these check valves because no performance problems were indicated from the maintenance history.
CV-22	RCIC check valve CV-22 (RCIC injection path discharge check valve) fails to open on demand. This valve must open for RCIC injection flow. The valve must also fully close when the pump is not in operation to prevent back-leakage and a possible waterhammer.	Detailed review completed.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
CV-2-27B	This valve is the feedwater isolation valve upstream of the RCIC injection path. The risk significant function of the component is to close to prevent RCIC from flowing back into the feedwater system. EPU uprate will increase the flow through this check valve by approximately 20%, however the function of the valve is not altered.	A detailed review was not performed for this check valve because no performance problems were indicated from the maintenance history.
CV-2-28B	Feedwater check valve CV-28B ('B' feedwater line check valve inside containment) fails to open on demand. This valve is located on drawing G-191167, H-5. Failure to open will prevent flow from either the RCIC or the Feedwater system. EPU uprate will increase the flow through this check valve by approximately 20%, however the function of the valve is not altered.	A detailed review was not performed for this check valve because no performance problems were indicated from the maintenance history.
CV-2-96A	Feedwater check valve V96A fails to open on demand. Failure of this valve will prevent flow from either the RCIC or the FW system. EPU uprate will increase the flow through this check valve by approximately 20%, however the function of the valve is not altered.	A detailed review was not performed for this check valve because no performance problems were indicated from the maintenance history.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
CV-40	RCIC check valve CV-40 (RCIC suction check valve from the suppression pool) fails to open on demand. This valve must open to provide a flow path from the torus to the RCIC pump suction.	A detailed review was not performed for this check valve because no performance problems were indicated from the maintenance history or walkdown.
CV-6/7	RCIC check valves CV- 6/7 (RCIC turbine exhaust check valves to torus) fails to open on demand.	Detailed review completed.
CV-72-109	Failure of check valve CV-109 (N2 bottle supply check valve to the plant N2 system) to close. The component is risk significant because if the check valve failed to close, the N2 bottle could bleed down to the plant N2 system.	Detailed review completed.
Digital Feedwater Control/Single Element Control	Following the modification that installed the digital feedwater control system, the licensee had problems with loss of inputs to the three-element controller (steam flow). This resulted in a reactor level transient. Since the event the plant had been operating in single-element control. Evaluate the modification and the acceptability of operating in single-element. Also determine if operation in single-element control would challenge the licensee's assumption that the plant would not scram following a single reactor feed pump trip, post-uprate.	Detailed review completed.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
DPIS-83/84	Spurious high steam flow signal. This steam flow instrument isolates RCIC steam in the event of a line rupture (indicated by high flow). Spurious isolation would result in the loss of RCIC flow.	These instruments are not included because there is significant margin in the setpoint to detect a steam line rupture, as well as margin between the normal operating point and the setpoint.
EOP/NPSH Fidelity	Verify fidelity between Emergency Operation Procedures and NPSH calculations and Containment Spray operation.	Detailed review completed.
FCV-2-4	FCV.4 (condensate pump minimum flow valve) fails to open on demand.	Detailed review completed.
FCV-2-4 Instrumentation	Failure of FCV.4 (condensate pump minimum flow valve) control instrumentation.	Detailed review completed.
Feed/Condensate Control	Operator fails to initiate and/or control feedwater/condensate.	Detailed review completed.
FT-58/FE-56	RCIC pump discharge flow instrument. This instrument is associated with the RCIC turbine control logic.	Detailed review completed.
GE SIL 351	GE SIL 351 - HPCI and RCIC Turbine Control System Calibration.	Vermont Yankee implemented SIL 351R.2 and provided the procedural changes recommended in the SIL for the HPCI system (OP 5337 Rev. 7). SIL 351 does not apply to RCIC since RCIC does not use a ramp generator (RGSC). This SIL is primarily procedural change recommendations and is not a high risk/low margin system.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
GE SIL 377	GE SIL 377 RCIC Startup Transient Improvement with Steam Bypass (June 24, 1982).	GE SIL 377 recommended a bypass for the steam supply line to the turbine for improved startup performance during a transient where RCIC is needed. This does not apply to Vermont Yankee since the SIL was a recommendation for plants who have issues with cold startup of the RCIC system. Upon talking to the system engineer, these issues have not existed for at least 20 years at VY.
GE SIL 467 (Bistable Vortexing)	GE SIL 467 and IEN 86-110 - Bistable vortexing is still a phenomenon that occurs periodically at VY.	The first occurrence of bistable vortexing at Vermont Yankee was following beginning of cycle 12 when recirculation system piping was replaced; however, this is a low risk event and thus does not meet the high risk / low margin criteria for this inspection. Vermont Yankee has had problems with bistable vortexing in the past and responded in depth to this SIL. The licensee responded to the SIL, added discussion on bistable vortexing at VY and action items for operators when bistable vortexing occurs. A review of Vermont Yankee's response to SIL 467, showed VY satisfied GE's recommended actions and placed guidance in OP 2110, Recirculation Procedure to aid the operators in identifying bistable vortexing.
GL 96-05, MOV Periodic Verification	GL 96-05 - Implementation of program for MOV Periodic Verification (As applicable to the selected sample of valves RCIC-MOV-15, 16, 131 and 132)	Detailed review completed.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
IN 2001-13 (SLC Relief Valve Margin)	Information Notice 2001-13 (8/10/01) - Inadequate Standby Liquid Control System Relief Valve Margin (Susquehanna, Units 1 and 2) Susquehanna's power uprate increased SRV setpoint pressure thus increasing SLC discharge pressure. However, the maximum SLC pump discharge pressure used a non-conservative maximum reactor vessel pressure in accident analysis.	Detailed review completed.
LER 3871995009 (LCO 3.0.3 Entry)	LER 1995-009-00 (7/3/95) - Condition Prohibited by the Plant's Technical Specifications (Susquehanna, Unit 1) - Non-conservative plant input into reactor core flow calculation.	Feedflow used in the analysis for power uprate is consistent with current feedflow indications.
LER 3251997005 (FW Indication Error)	LER 1997-005-01 (8/8/97) - Feedwater Flow Indication Discrepancy (Brunswick Steam Electric Plant, Unit 1).	Vermont Yankee does not have and is not required to have chemical tracer mass flow rate tests. This is more conservative then having the tracers since the chemical tracer mass flow rate tests are controversial and have had past issues. VY is waiting for industry or regulatory guidance on this issue before adding this test.
LER 2961998001 (LOCA Sensor Problem)	LER 1998-001-00 (4/1/1998) - Computer Modeling Indicates Sensors May Not Detect All Possible Break Locations (Browns Ferry, Unit 3).	Vermont Yankee does use the GOTHIC computer code to analyze high energy pipe breaks; however, this is a low risk issue and presented no significant safety issue at Browns Ferry.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
LER 2601999009 (Scram Due to EHC Leak)	LER 1999-009-00 (10/14/99) - Manual Reactor Scram Due to EHC Leak (Browns Ferry Nuclear Power Station, Unit 2).	The EHC leak was on a very specific 3/8 inch nominal outer diameter tubing connection which consisted of socket weld glands and standard nuts to connect the accumulator to a pressure transmitter. The leak was due to poor fabrication and poor work practices specific to Browns Ferry.
LER 2372001005 (1/7/02)	LER 2001-005-00 (1/7/02) - Unit 2 Scram Due to Increased First Stage Turbine Pressure (Dresden, Unit 2).	Vermont Yankee responded to GE SIL 423, in 1998, by implementing corrective actions.
LER 4612002002 (Inadequate PM on FW System)	LER 2002-002-00 (7/11/02) - Inadequate Preventive Maintenance Program for the Feedwater System Results in Lockup of a Turbine-Driven Reactor Feed Pump and Scram on High Reactor Pressure Vessel Water Level During Extended Power Uprate Testing (Clinton Power Station). Feedwater increased due to the power uprate; however, the feedwater limit switch did not increase to accommodate this increase in flow.	This operating experience does not apply since Vermont Yankee does not have turbine driven feedwater pumps, and this issue does not apply to other turbine driven pumps in the plant.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
LER 3412002005 (Non-Conservative Setpoint)	LER 2002-05 (1/16/03) - Discovery of Non-Conservative Setpoint for the Thermal-Hydraulic Stability Option III Oscillation Power Range Monitor (OPRM) Period Based Algorithm, Tmin (Fermi, Unit 2).	This OE does not apply to Vermont Yankee since power oscillations are monitored using approved BWROG Option 1D not Option III. Vermont Yankee does not have Oscillation Power Range Monitors, Period Based Detection Algorithms, and Tmin values. Option III is used for larger BWRs that have local power oscillations. Since Vermont Yankee has a small BWR core, only core-wide oscillations occur (not local oscillations). The inspector met with an individual from power uprate (and used to work in reactor engineering) and discussed, in detail, core monitoring using Option 1D for the new ARTS/MELLA core design and the power uprate core design.
LER 4542003003 (Maximum Power Exceeded)	LER 2003-003-00 (9/29/03) - Licensed Maximum Power Level Exceeded Due to Inaccuracies in Feedwater Ultrasonic Flow Measurements Caused by Signal Noise Contamination (Byron).	Detailed review completed.
LER 3411992009	LER-92-009-00 (11/20/92) - Safety Relief Valves Set Pressure Outside Technical Specifications (Fermi, Unit 2).	VY has had no issues with setpoint drift on the SRVs or RVs in containment. Setpoint drift considered in this LER was an indication of disc-to-seat sticking due to corrosion binding on the SRVs and RVs at Fermi thus making these valves fail their set pressures tests.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
LSHH-4A	Level switch LSHH 4A contacts fail/short. High Water Make up - Condenser level Control Switch Fails high - auto make malfunctions to the CST - Operator Action is required. No EPU impact.	Operator can take manual action to overcome this failure. The consequence of the failure of the switch is not significant because the operator can take manual control.
Manual Initiation of HPCI/RCIC	Operator fails to manually initiate HPCI and RCIC systems.	Detailed review completed.
Manual Operation of SRVs (Medium LOCA)	Operator fails to manually open the SRVs for a medium LOCA.	Emergency Operating Procedures (EOP) require operator action to manually open the SRVs to depressurize the reactor under medium break LOCA conditions. Validation studies and operator observations in the simulator have shown that given various factors that influence human performance (stress, training, equipment failures, etc.), the task to open the SRVs manually would be accomplished in less than 7 minutes which is lower than the 33 minutes (or 24 minutes for CPPU) needed to assure > 1/3 core coverage. Additionally, operator training frequently focuses on this event making it unlikely that the operator would fail to perform the task within the required time.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
Manual Operation of SRVs (Small LOCA/Transient)	Operator fails to manually open the SRVs for transient/small LOCA.	Emergency Operating Procedures (EOP) require operator action to manually open the SRVs to depressurize the reactor under transient and small break LOCA conditions. Validation studies and operator observations in the simulator have shown that given various factors that influence human performance (stress, training, equipment failures, etc.), the task to open the SRVs manually would be accomplished in less than 5 minutes which is much lower than the 66 minutes (or 48 minutes for CPPU) needed to assure > 1/3 core coverage. Additionally, operator training frequently focuses on this event making it unlikely that the operator would fail to perform the task within the required time.
Manual RCIC operation-Appendix R Safe Shutdown	Appendix R Safe Shutdown Analysis - Operator fails to manually initiate RCIC system using alternate shutdown panels (Generic Human Actions that are Risk Important), and GE document NEDC-330090P, Table 10-5 (Assessment of Key Operator Action).	Detailed review completed.
MOV-131	RCIC MOV 131 (RCIC turbine steam supply valve) fails to open on demand. This valve is required to open to provide steam to the RCIC turbine for operation.	Not included because valve has adequate design margin to open when required.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
MOV-132	RCIC MOV 132 (cooling water valve to the RCIC lube oil cooler) fails to open on demand. This valve is required to open to provide cooling water to the RCIC pump lube oil cooler. Failure to cool the lube oil could result in failure of the pump/turbine.	Not included because valve has adequate design margin to open when required.
MOV-15/16	RCIC MOV 15/16 (steam supply to RCIC turbine) fails closed during its mission time. These valves are required to close in the event of a line break in the RCIC turbine steam supply to isolate the HELB. These valves are also required to remain open when the RCIC pump is required to operate.	Detailed review completed.
MOV-18	RCIC MOV 18 (RCIC pump suction valve from the CST) transfers closed during its mission time. This valve is required to automatically close when the RCIC pump suction is transferred from the CST to the torus. This valve must remain open while the RCIC pump is operating from the CST.	Not included because valve has adequate design margin to close when required.
MOV-21/20	RCIC MOV 21 (inboard discharge valve to the reactor vessel) fails to open on demand. Also look at MOV-20 (the normally open outboard discharge isolation valve). These valves must automatically open to provide RCIC injection flow in response to an RCIC initiation signal.	Detailed review completed.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
MOV-27	This is the RCIC minimum flow valve. This valve is required to open at low RCIC flow to protect the pump.	Detailed review completed.
MOV-39	RCIC MOV 39 (RCIC suction valve from the suppression pool) fails to open on demand. This valve is required to open when the RCIC pump suction is transferred from the CST to the torus.	Detailed review completed.
MOV-41	RCIC MOV 41 (RCIC suction valve from the suppression pool) fails to open on demand. This valve is required to open when the RCIC pump suction is transferred from the CST to the torus.	Not included because valve has adequate design margin to open when required.
MOV-64-31	MOV 64-31 (manual makeup valve from the CST to hotwell) fails to open on demand.	Failure of this valve will prevent make-up from the hot-well to the CST. The loss of this valve would not be safety significant and there are no indications that there is low margin on for this valve
Offsite Transmission System	Offsite Transmission System: preferred source of power to the 4160V safety buses; must remain stable and available following the trip of the VY generator.	Detailed review completed.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
Operator Bypasses the MSIV Isolation Interlocks	Operator Bypasses MSIV Isolation Interlocks. The justification is the decrease in the Allowable Action Time for the operators at the EPU level (CPPU). It is based on input from the Human Performance technical staff, Appendix A of NUREG 1764 (Generic Human Actions that are Risk Important), and GE document NEDC-330090P, Table 10-5 (Assessment of Key Operator Action).	The allowable action time to bypass the MSIV low-low level isolation interlocks is based upon the time it would take to reach the RPV low-low level setpoint for an ATWS with no injection. Validation studies by the licensee have shown that the task would be accomplished for transient and LOCA events within the required time. The margin to accomplish the task is adequate, for current and CPPU conditions, given other operational factors and steps in the EOPs which must be taken into account (e.g., a high main steam line radiation isolation signal maintaining the valves closed). Operators train and are evaluated and tested on a regular basis for this scenario further reducing the likelihood that the task would not be completed in the time required.
Operator Inhibits ADS	Operator action to inhibit ADS. The justification is the decrease in the Allowable Action Time for the operators at the EPU level (CPPU). It is based on input from the Human Performance technical staff, Appendix A of NUREG 1764 (Generic Human Actions that are Risk Important), and GE document NEDC-330090P, Table 10-5 (Assessment of Key Operator Action).	The operator action to inhibit ADS is one of the first actions taken by the operators under certain transient conditions in the EOPs. The allowable action time is based on the time to reach the vessel level low-low set point for ATWS without injection plus two minutes for the ADS timer. Validation studies and operator observation in the control room have demonstrated that the action would be accomplished in less than 3 minutes. The margin to complete the task is not significantly changed under CPPU conditions. Additionally, operators are trained and tested regularly in this EOP action step.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
Passive Failure of Feedwater Piping	Review effect of increased feedwater flow on flow-accelerated corrosion rates following the power uprate.	Detailed review completed.
PB IR 2002-011 (HPCI Functional Issue)	Peach Bottom Finding for IR 50-277/2002-011 (8/5/02) - Finding Related to High Pressure Coolant Injection Function (may apply to RCIC system at VY).	Detailed review completed.
PCV-23	RCIC PCV 23 (RCIC air operated lube oil temperature control valve) fails to open on demand. This valve uses instrument air to control its setpoint and fails fully open on a loss of instrument air. This valve is required to provide cooling water, at the correct pressure, to the RCIC pump lube oil cooler when the RCIC pump is operating.	Detailed review completed.
PS-67	Spurious RCIC low suction pressure trip signal. This instrument will cause the RCIC pump to trip in the event of low pump suction pressure. Spurious trips will result in a loss of RCIC flow.	Not included because there is significant margin in the setpoint to prevent a spurious trip.
PSH-72A/B	Spurious RCIC turbine exhaust high pressure trip. This instrument will trip the RCIC pump in the event of high pressure in the exhaust steam line. Spurious trips will result in a loss of RCIC flow.	Not included because there is significant margin in the setpoint to prevent a spurious trip.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
PT-59/60	RCIC pump discharge pressure. This instrument is associated with the RCIC turbine control logic.	Not included because there is significant margin in the setpoint.
PT-68	Spurious low steam line pressure signal. This instrument will isolate steam flow to the RCIC turbine in the event of low steam supply pressure, indicating a steam line break. Spurious isolation would result in a loss of RCIC flow.	Not included because the pressure switch setpoint has significant margin to prevent a spurious pump trip.
PT-70	Spurious RCIC trip on high turbine exhaust pressure signal. Component ID is PT-70. Include exhaust rupture disks S3 and S4. This instrument will trip the RCIC pump in the event of high pressure in the exhaust steam line. Spurious trips will result in a loss of RCIC flow.	Not included because there is significant margin in the setpoint and operating pressure to prevent a spurious trip.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
Manual operation of MOV 64-31	Operator fails to manually open MOV 64-31 (used to manually transfer makeup from the CST to the condenser).	The operator action to manually open valve MOV 64-31, Hotwell Emergency Makeup Valve, is performed in the main control room. The action is required when turbine bypass is not available (during an MSIV closure event). In that case automatic makeup to the hotwell from the Condensate Storage Tank (CST) may not be sufficient to keep up with reactor vessel makeup requirements (feedwater pumps providing vessel level makeup). Validation studies and operator observations have estimated a 1 minute time to manipulate the valve from the control room. If the valve is required to be opened from the field the estimates are less than 15 minutes, however, other EOP mitigation strategies such as use of low pressure ECCS pumps, would assure core coverage if the valve could not be opened.
RB/Torus Vacuum Breakers	Reactor Building to Torus vacuum breakers. The vacuum breakers are required to open to prevent a vacuum in the containment. These also must remain closed to ensure containment integrity and to prevent loss of overpressure for ECCS NPSH.	Detailed review completed.
RCIC Pump P-47-1A and Turbine TU-2-1-A	RCIC pump P-47-1A fails to start on demand. This sample includes the turbine driven RCIC pump, the governor valve, and trip throttle valve.	Detailed review completed.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
Reactor Feed Pump	Failure of the feedwater pump will fail to deliver flow required for normal operation or to mitigate an accident. Prior to EPU 2 of three feedwater pumps are required to support the Feedwater system requirements. As such there is a 50% spare capability. For EPU three pumps are required to operated due to the increase requirements of feedwater flow.	Detailed review completed.
RHR Pump	Review RHR pump NPSH calculation, associated suction strainers, bubble ingestion, and torus vortexing issues.	Detailed review completed.
Safety Valve (New)	Addition of third main steam safety valve for power uprate. Failure of SSV to open and relieve pressure during transients or small/medium break LOCA.	Detailed review completed.
SLC Initiation with Condenser Failed	Operator fails to initiate SLC with the main condenser failed. The justification is the decrease in the Allowable Action Time for the operators at the EPU level (CPPU). It is based on input from the Human Performance technical staff, Appendix A of NUREG 1764 (Generic Human Actions that are Risk Important), and GE document NEDC-330090P, Table 10-5 (Assessment of Key Operator Action).	Detailed review completed.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
Spurious High Steam Line Space Temperature Trip	Spurious RCIC trip on high steam line space temperature (instrument TS 79 through 82). These instruments would result in isolation of the steam flow to the RCIC turbine in the event of a steam line break. A spurious trip would result in loss of RCIC flow.	Not included because there is significant margin between the setpoint and the operating temperature to prevent a spurious trip.
Spurious High Steam Tunnel Temperature Trip	Spurious RCIC trip on a high steam tunnel temperature trip signal. These instruments would result in isolation of the steam flow to the RCIC turbine in the event of a steam line break. A spurious trip would result in loss of RCIC flow.	Not included because there is significant margin between the setpoint and the operating temperature to prevent a spurious trip.
Spurious Reactor High Level Trip	Spurious high reactor water level signal (trip could affect both the RCIC pump or feed water pump). These instruments would result in tripping the RCIC turbine in the event of high RPV level. A spurious trip would result in loss of RCIC flow.	Excluded because HPCI and the RFP trip signals are provided by different instruments and the probability of a simultaneous failure of these instruments is extremely low.
SR-26	SR-26 (RCIC supply to lube oil cooler relief valve) fails open. This component is designed to protect the RCIC lube oil cooler and may be important on a loss of IA when the flow control valve fully opens (based on interview with RCIC System Manager).	Detailed review completed.
SRVs	Safety relief valves allow the reactor to be depressurized.	Detailed review completed.

SSC/OA/OE	Description	Detailed Review Completed / Basis For Exclusion
Vernon Tie Line	Operator monitoring of Vernon tie line to ensure availability as a station blackout source.	Detailed review completed.

March 28, 2006

The Honorable Hillary Rodham Clinton
United States Senate
Washington, D.C. 20515

Dear Senator Clinton:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your request during the hearing of the Senate Environment and Public Works Committee, Subcommittee on Clean Air, Climate Change, and Nuclear Safety, on March 9, 2006, for an independent safety assessment to be conducted of the Indian Point Energy Center, which is operated by Entergy Nuclear Operations, Inc. The Commission understands your desire for independent, thorough, and objective inspections. The NRC is an independent regulatory agency and our established inspection and assessment processes are independent, thorough, and objective.

As explained at the hearing, the inspections and assessments carried out through the NRC's current Reactor Oversight Process (ROP), now in its seventh year of implementation, have been greatly enhanced, in part, by incorporating the insights gained from past inspections and assessments, such as the assessment performed at Maine Yankee in 1996. The Maine Yankee Independent Safety Assessment (ISA) was a unique, one-time review, and the Commission does not believe that an effort to replicate a "Maine Yankee" ISA is warranted.

The ROP inspection plan now includes the performance of an extensive engineering team inspection at each reactor facility every two years. Recently, the engineering team inspection effort was significantly enhanced to provide for a more effective review of the plant design and operational configuration of components that are important to safety. The engineering team inspection is performed by a multi-disciplinary team consisting of NRC inspectors as well as outside contractors and is scheduled to be performed at both Indian Point Units 2 and 3 in 2007. The NRC inspects Indian Point Units 2 and 3 separately, in effect doubling inspection scrutiny for key inspections such as this engineering inspection. For each unit, the inspection is expected to last seven weeks, including four weeks of on-site time, and involve approximately 700 hours of direct inspection effort. As with all NRC inspections at Indian Point, representatives from the New York State government are welcome to observe or participate. In performing this inspection, the NRC staff will use operating experience, risk assessment, and engineering analysis to select safety-significant components and operator actions to verify that the selected components are capable of performing their intended safety function and that operating procedures are consistent with the design and licensing bases. The combination of the risk-informed overall baseline inspections with the improved engineering-focused inspection provides a more safety-focused review of significant plant components. This improved engineering team inspection was pilot-tested at four sites, including Vermont Yankee in the Fall of 2004, and proved to be an excellent inspection process.

In addition, the NRC resident inspectors for Indian Point conduct routine inspections on a continuing basis. NRC regulations and oversight process focus on ensuring nuclear safety and security across seven “corner stone” areas, including emergency planning. Under our current ROP, NRC resident inspectors and regional specialists with specific areas of expertise routinely sample and evaluate the work performed by Entergy’s engineering organization to determine whether engineering analyses adequately support safe operation.

The Commission believes that the current increased level of oversight at Indian Point is appropriate and that the scope and depth of the ROP inspection activities, including the enhanced engineering team inspections, are superior to the processes in place at the time the decision to conduct the Maine Yankee assessment was made. Based on this, we believe that the performance of this inspection regimen will effectively accomplish the intent and objectives of the assessment you discussed at the meeting. If you have further questions or would like a briefing on these issues, please contact me.

Sincerely,

/RA/

Nils J. Diaz

November 1, 2005

Dr. Robert Claypool
Office of Mass Casualty Planning
Office of Public Health Emergency Preparedness
U.S. Department of Health and Human Services
200 Independence Avenue SW., Room 638G
Washington, D.C. 20201

Dear Dr. Claypool:

On behalf of the Nuclear Regulatory Commission (NRC), I am providing the following comments on the draft Federal guidelines to make potassium iodide (KI) available to jurisdictions within a 20-mile radius of nuclear power plants. The Federal Register notice (FRN) that promulgated the draft guidelines also requested comments on whether the expanded distribution of KI was necessary, considering existing preventive measures and/or other thyroid prophylaxis.

The draft guidelines provide a good discussion of potassium iodide, the radiological emergency planning efforts for commercial nuclear power plants, the potential consequences of terrorism on nuclear power plants, and the impacts of the 1986 Chernobyl accident. The guidelines would provide to State, local, and tribal governments a framework for considering whether to expand distribution of KI out to 20 miles around nuclear power plants. While Section 127 of P.L. 107-188, the Public Health Security and Bioterrorism Preparedness and Response Act of 2002 (the Bioterrorism Act), refers to distribution of KI *tablets*, the proposed guidance generally refers to distribution of KI.

The NRC staff does not have specific comments on the draft guidelines. However, the NRC staff concludes that the predetermined protective actions in place for the populations within the 10 and 50 mile Emergency Planning Zones provide the necessary protection for the thyroid gland from radioactive iodine and that expanded distribution of KI is unnecessary.

Expanded distribution of KI could negatively impact the current, well-established, and scientifically sound framework of the NRC's emergency preparedness regulations. The NRC and the Federal Emergency Management Agency regulatory framework for emergency preparedness was put into place after the 1979 accident at Three Mile Island Unit 2. Each nuclear power plant operator was required to submit the radiological emergency response plans of State and local governments that are within the 10-mile plume exposure pathway emergency planning zones (EPZ), as well as the plans of State governments within the 50-mile ingestion pathway EPZs. These emergency planning zones facilitate the implementation of a preplanned strategy for protective actions during an emergency.

As the draft guidelines point out, NRC analyses indicate that, in the event of an emergency at a nuclear power plant that causes a release of radioactive materials in excess of routine low-level effluents, exposure to these materials poses the greatest risk for people closest to the plant. The risks to these people would arise from the exposure pathways of direct shine, immersion in a plume, inhalation and ingestion of radioactive materials, and ground shine. The objectives of the predetermined protective actions within the 10-mile EPZ, which include sheltering, evacuation, and, where appropriate, the use of potassium iodide, are to mitigate these risks in the event of an emergency.

The population at greater distances from the plant may be at risk of exposure to radioactive materials by way of ingestion of these materials. Predetermined protective actions for the 50-mile ingestion exposure pathway EPZ include interdiction of contaminated milk, food, and water as well as protective measures for livestock.

Section 127 of the Bioterrorism Act directed the National Academy of Sciences (NAS) to study the expanded distribution of potassium iodide and report back to the President on the best distribution methods to accomplish such an expanded distribution. The NAS published this study in January 2004. Although the NAS did not identify any one particular “best method” of distribution, the Academy raised questions regarding the usefulness of expanded distribution of KI. Specifically, Chapter 5 of the report states (on page 81): “Exposure to radioactive iodine is possible through the ingestion pathway, so it is important that plans address this situation. Monitoring of the environment and food products controls this route of exposure. Removing contaminated products from the market and isolating contaminated products until the radioactive iodine decays to safe levels are the most effective way to eliminate radiation exposure and damage to the thyroid. That also eliminates the need for the use of KI by the general public as a protective action.” In the conclusions and recommendations of the NAS report (on page 159), the Academy summarized this finding as follows: “KI is also effective for protection against the harmful thyroid effects of radioiodine ingested in contaminated milk and other foods, but food testing and interdiction programs in place throughout the United States are more effective preventive strategies for ingestion pathways.”

These NAS findings have been buttressed by the most recent report of the International Atomic Energy Agency’s (IAEA’s) Chernobyl Forum on the health effects of the Chernobyl accident, which was issued in August 2005. This report included a finding that ingestion of contaminated milk products was the primary cause of the thyroid cancers found in children living in the surrounding regions. Consequently, interdiction of contaminated milk and use of stored feed would have prevented most of the thyroid cancers found in these children.

Therefore, we have concluded that other, more effective, protective measures are in place to protect the thyroid gland in the event of a release of radioactive iodine, and that expanded distribution of KI is unnecessary. Thus, the NRC recommends that the Secretary of Health and Human Services, as delegated by the President, apply subsection 127(f) of the Bioterrorism Act.

Thank you for the opportunity to comment on these important guidelines. If you have any questions or would like to discuss our comments, please do not hesitate to contact Eric Leeds, the NRC's Director of Preparedness and Response, at 301-415-2334.

Sincerely,

/RA/

William F. Kane
Deputy Executive Director for Reactor
and Preparedness Programs
Office of the Executive Director for Operations

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Preliminary Listing of Events Involving Tritium Leaks
March 28, 2006

The following presents a preliminary listing of events that have involved tritium leaks and spills at nuclear power reactor sites. This information is being compiled as part of several issues being addressed by the NRC's Liquid Radioactive Release Lessons-Learned Task Force. Specifically, the Task Force will conduct a review of inadvertent releases (from 1996 to present) of radioactive liquid to the environment, obtain additional information from each reactor site to complete its evaluation, and prepare recommendations focusing on specific improvements regarding operational practices and the NRC reactor inspection program. The information will be included in a report expected to be released in September 2006. Accordingly, the information presented at this time is preliminary and will be subject to revision as new or supplemental information is obtained by the Task Force.

To put all of these leaks into some perspective, all operating commercial nuclear power plants release tritiated water. That is part of their licensing basis. Typical releases are on the order of hundreds of curies per year, although they can be in thousands of curies per year and still meet strict regulatory requirements.

The EPA regulatory limit for tritium in drinking water is 20,000 picocuries (a millionth of a millionth of a curie) per liter. If a person drinks 2 liters per day of such water every day for a year, the person would receive about 4 mrem of radiation dose. That is about the same dose as would be received from eating a banana a day or flying from Washington to Los Angeles or spending 100 hours in the Capitol.

1. NRC Region I Plants

Haddam Neck, CT

In October 2005, the Haddam Neck plant identified soil contamination near an exterior spent fuel pool wall. The plant ceased operations about ten years ago and is being dismantled under an approved decommissioning and license termination plan. Trace indications on the wall indicated potential previous pool leakage. The soil may have been contaminated by a past spent fuel pool leak or leakage from radioactive water storage tanks which was previously identified and corrected by the licensee. As of June 2005, tritium levels have ranged from about 300 to 4,000 picocuries per liter as a result of ongoing remediation activities. Earlier results, have revealed concentrations on the order of 30,000 picocuries per liter onsite. The licensee is also evaluating ground water samples for the presence of other radionuclides, such as strontium 90. No offsite public dose impacts have been identified. The licensee is continuing to evaluate and characterize the extent of onsite contamination as part of demolition and remediation activities.

Indian Point - 2, NY

In September 2005, the utility discovered a small amount of water leaking from the spent fuel pool and the subsequent discovery of subsurface ground water contamination in a monitoring well located on site within the transformer yard. Tritium concentrations were noted to range from about 500,000 picocuries per liter of water in the immediate vicinity of the leak, to about

30,000 picocuries per liter near the discharge canal, and about 1,000 picocuries per liter near the outfall of the discharge canal. A bounding calculation estimated that the release of radioactivity from ground water into the nearby Hudson River would result in a dose of 0.0001 mrem per year to the maximally exposed individual based on a leak rate of less than 3.0 gallons per day out of the spent fuel pool. The resulting dose was found to well below the NRC-required dose limit of 3 mrem per year specified in NRC regulations and by plant technical specifications. NRC is pursuing independent sampling and analysis of well samples which have been sampled by the licensee. The NRC issued a report presenting the results of its inspections and findings on March 16, 2006.

Salem - 1, NJ

In 2003, the licensee identified tritium adjacent to the Salem Unit 1 spent fuel pool within the restricted area. Specifically, water leaked from the Unit 1 spent fuel pool for an undetermined period of time through December 2002 and accumulated between the spent fuel pool liner and fuel building wall. The water subsequently leaked through the building walls presenting the potential for undetected releases of contaminated water. The highest level of tritium in ground water was estimated at about 3.5 million picocuries per liter at the test location. This event was the subject of an NRC special inspection conducted in August 2003. The State of New Jersey was notified under NRC regulations. To date, there is no evidence that tritium contaminated ground water has been released into the public domain beyond the site boundary. The licensee installed a number of monitoring wells and is pumping out contaminated ground water for processing as radioactive waste. The NRC is monitoring the licensee's activities. In addition, the State of New Jersey has been involved in sampling and monitoring of the site wells.

Seabrook, NH

Seabrook has experienced some leakage out of the fuel transfer canal and fuel cask handling area since 1999. From 1999 to mid 2002, the leak rate ranged from 0.01 to 1 gallon per day. In mid 2002 to April 2004, the leak rate increased to approximately 30 to 40 gallons per day. During the period of April 16 to 19, 2004, in an effort to specifically determine the location of the leak, the leak rate increased to approximately 350 gallons per day. Following the discovery of this increase, the plant drained the fuel transfer canal, stopping the leakage. Subsequently, Seabrook conducted several inspections of the fuel transfer canal and cask handling areas and identified a crack, approximately 2 foot by 1/8 inch, in a weld between a support steel plate and canal liner. This was identified as the source of the leak. An NRC environmental inspection, conducted in December 2005, found the fuel transfer canal repairs were effective. The leak rate was found to have stabilized at approximately 0.02 gallon per day.

The licensee installed several ground water monitoring wells in June 2004 to determine tritium migration at the site. The licensee has since monitored the various onsite wells to evaluate the movement of tritium into the environment. Tritium levels measured outside of the buildings where leaks were noted were found to remain well within NRC regulatory limits and within the EPA drinking water limit of 20,000 picocuries per liter. Onsite monitoring wells showed tritium levels near non-detectable levels, indicating that tritium levels were associated with the old leakage. In addition, no tritium was detected in other environmental samples and no other related radionuclides were detected in similar environmental samples. For 2004, the maximum dose to a hypothetical individual was estimated to be about 0.03 mrem from all exposure pathways. This dose is below the NRC As Low as is Reasonably Achievable (ALARA) objective of maintaining the dose below 3 mrem per year, specified in Appendix I to 10 CFR Part 50. This is also well below the NRC's 100 mrem per year radiation safety limit in 10 CFR Part 20.

Oyster Creek, NJ

The radioactive condensate liquid system, cross-connected to a cooling water system, resulted in the discharge of about 130,000 gallons of radioactive liquid to the discharge canal and then to Barnegat Bay in September 1996. The total radioactivity release was estimated to be about 7.5 curies and consisted mostly of tritium.

2. NRC Region II Plants

Watts Bar, TN

In August 2002, low levels of tritium (less than 1,000 picocuries per liter) were detected in one onsite well. The well is sampled as part of the routine radiological environmental monitoring program. Additional sampling conducted in 2003, revealed the presence of tritium in three other wells, at levels up to 20,000 picocuries per liter. Historically, concentrations of tritium have been about 5,000 picocuries per liter, but in January 2005, there was a sudden increase to about 500,000 picocuries per liter. Several potential sources of leakage were identified, including liquid effluent lines, fuel transfer canal and tube, refueling water storage tank, spent fuel pool, and the spent fuel cask loading pit. In addition, numerous liquid process system tanks were inspected for evidence of leakage. The licensee has taken actions to reduce or eliminate the identified sources of leakage into ground water. In addition, the licensee is continuing to monitor levels of tritium in ground water to assess the movement and extent of ground water contamination. The NRC has scheduled an inspection during which these issues will be reviewed and evaluated.

3. NRC Region III Plants

Braidwood, IL

On November 30, 2005, the licensee informed the NRC Resident Inspectors of higher than expected tritium levels (about 60,000 picocuries per liter) measured in onsite monitoring wells at the northern edge of the owner controlled area. More recent sampling and analysis revealed tritium levels on the order of several hundred thousands of picocuries per liter in ground water and, recently, similar levels in water spilled in an area where contaminated water is being stored in tanks. The licensee attributed the higher levels of tritium to historical vacuum breaker valve leakage in the circulating water blowdown line to the Kankakee River that occurred in 1998 and 2000. The licensee uses the blowdown line to conduct liquid effluent releases to the river.

Recently, the licensee has detected measurable levels of tritium in offsite ground water. Elevated levels were detected in one offsite monitoring well in a vacant development (on the order of 30,000 picocuries per liter). One nearby residential well was found to have detectable tritium (about 1,500 picocuries per liter). The licensee is continuing to sample and to develop plans for remediation. The licensee has issued a report identifying the cause of the spills and is working with the NRC on expanding its surface and ground water sampling and analysis program. NRC is also pursuing independent sampling and analysis of well. The NRC is expected to issue a report outlining the results of its inspections and findings in the near future.

Byron, IL

As a result of the issue identified at Braidwood and subsequent NRC inspections, at Byron the licensee initiated a sampling and analysis program along its discharge line to the Rock River. All valve pits have been inspected, and 5 of the 6 pits were identified to have some standing water and levels of tritium. The levels ranged from just above detectable to about 80,000 picocuries per liter. The licensee suspended all radioactive liquid effluent releases. Additional well installations were planned in February 2006. Two residential wells have been sampled, both have shown negative results. The licensee is pursuing sampling and analysis of an additional nine residential wells. The NRC plans to accompany the licensee during their residential sampling and offer independent NRC sampling and analysis. NRC will also pursue independent sampling and analysis of the two wells which have been sampled by the licensee.

Callaway, MO

A January 2005 pipe break, due to onsite construction activities, resulted in the contamination of soil and ground water. All effluent discharges were suspended until the pipe was fixed, soil and water samples were taken and evaluations were made. The pipe break was located beyond where the radiation monitor is situated on the discharge pipe. As a result, the release was monitored at all times during the discharge and, consequently, the radioactive material and associated dose were accounted for in the permit allowing the release. The licensee has included the location around the pipe break for follow up sampling and analysis as part of its radiological environmental monitoring program.

Dresden, IL

In August of 2004, the licensee identified an underground leak of its condensate storage tank (CST) piping. The licensee detected levels of tritium in onsite ground water monitoring wells as high as 1,700,000 picocuries per liter, with current levels about 600,000 picocuries per liter. Onsite tritium levels in the two closest wells have stabilized at about 20,000 to 50,000 picocuries per liter. The licensee isolated the leakage and replaced the faulty section of piping in November 2004. Onsite monitoring well data confirm that the flow of groundwater is generally away from residential areas and towards the river. In 2004 and 2005, the licensee sampled the private wells of nearby residents. One of the residents' wells had measurable levels of tritium above background (approximately 1,000 picocuries per liter) and has shown positive results for tritium for a number of years. However, the licensee's other monitoring results and an independent hydrology study do not appear to support that the elevated levels of tritium in that well were from the 2004 CST pipe leakage. The licensee continues to evaluate the tritium in that well, which is a normal sample point for its radiological environmental monitoring program. The NRC is following up on the licensee's actions.

4. NRC Region IV Plants

Palo Verde, AZ

The licensee has reported that tritium has been detected at the Palo Verde site. The licensee dug a hole about 13 feet deep near the Unit 3 tunnel where the spray pond piping penetrates the vault, in response to water found inside the pipe tunnel. The licensee took three samples of water and tested them for tritium. The licensee's testing found tritium in all three samples; at 75,000, 30,000, and 70,000 picocuries per liter. At this time, the licensee has identified three potential sources for the tritium, including a holdup tank, nearby system piping, and from plant stack discharges that were washed into the ground during rainfall. NRC's Region IV inspectors are evaluating the licensee's effort in tracing the origin of the leak and in quantifying the amounts of tritium and its movement in ground water.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

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NRC CREATES TASK FORCE TO EXAMINE TRITIUM ISSUE

The Nuclear Regulatory Commission said today it has assembled a group of experts from its offices around the nation to examine the issue of inadvertent, unmonitored releases of radioactive liquids containing tritium from U.S. commercial nuclear power plants.

Agency leaders directed creation of the group earlier this year following reports of unmonitored releases of water containing tritium.

"The available information on these releases shows no hazard to the public," said NRC Executive Director for Operations Luis Reyes. "Nonetheless, we need to conduct an in-depth review to see if the NRC needs to take additional action of a broad nature."

At the same time the NRC decided to establish the tritium study group, they also decided to create a page on the NRC Web site to provide the public the latest available information on tritium issues. This information can be accessed at this address: http://www.nrc.gov/reactors/operating/ops-experience/grndwtr-contam-tritium.html .

Eleven of the 12 task force members come from the agency's Offices of Nuclear Reactor Regulation, Nuclear Material Safety and Safeguards and Nuclear Regulatory Research, as well as from regional offices. The twelfth, a representative of state government, is being selected. The group will report to Bill Kane, the Deputy Executive Director for Reactor and Preparedness Programs, and is required to complete its review by Aug. 31. A written report summarizing the task force's findings will be issued late this year.

The task force is required to address several topics, including:

– A general assessment of the potential public health impact from these releases;
– How the issue was communicated to the public, state and local officials, other federal agencies, Congress and other interested groups;
– A review of other inadvertent releases at nuclear power plants, including decommissioning sites, from 1996 to the present;
– Industry actions in response to the releases, including the timing of remediation efforts; and,
– NRC oversight of inadvertent releases, both under the Reactor Oversight Process (ROP) and the process in place prior to the ROP.

The task force can also consider issues not listed in its charter, and can identify issues for longer-term review by NRC staff.

The task force's charter is available on the NRC's Web site by entering ML060690186 at this address: http://adamswebsearch.nrc.gov/dologin.htm .

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