NEI 99-03

Control Room Habitability Assessment Guidance

NUCLEAR ENERGY INSTITUTE

June 2001

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ACKNOWLEDGEMENTS

The Nuclear Energy Institute (NEI) Task Force on Control Room Habitability developed the *Control Room Habitability Assessment Guidance* document. We appreciate the task force members contributing to its development and the industry contributors who commented on it to improve the content and clarity.

EXECUTIVE SUMMARY

The process described in NEI 99-03 is designed to ensure that the licensing and design bases associated with control room habitability (CRH) are verified and maintained. The document provides a comprehensive set of tools that function together to improve overall CRH program performance. The document's guidance facilitates adequate protection of control room operators against the effects of postulated releases of radioactive particulates or gases, toxic gas or external smoke. It also guides the development of a Control Room Envelope Integrity Program to provide long-term maintenance of the control room envelope. In summary, its elements define an acceptable method to demonstrate and maintain CRH.

The document is divided into three parts:

- Background
- Assessment Process
- Establishment and Maintenance of Control Room Envelope (CRE) Integrity

Part 1, *Background*, provides basic information related to CRH, establishing a foundation from which the CRH assessment can proceed.

Part 2, *Assessment Process*, determines if the plant configuration and operation are consistent with the CRH licensing basis and analysis. Guidance is provided on:

- Identifying the current plant licensing and design bases for comparison with actual plant configuration and operation
- Determining the applicability and recommended actions for generic concerns associated with:
  - Analysis of the limiting design basis accident
  - Toxic gas events
  - External smoke events
- Performing a baseline unfiltered control room envelope in-leakage test to determine conformance with licensee assumed in-leakage values
- Managing degraded conditions.

Part 3, *Establishment and Maintenance of CRE Integrity*, guides development of a licensee controlled program to manage CRH after the evaluations of Part 2 are completed. The recommended program includes periodic assessment of in-leakage and maintenance of CRE integrity.

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PART 1 - BACKGROUND

1 INTRODUCTION

1.1 PURPOSE AND SCOPE

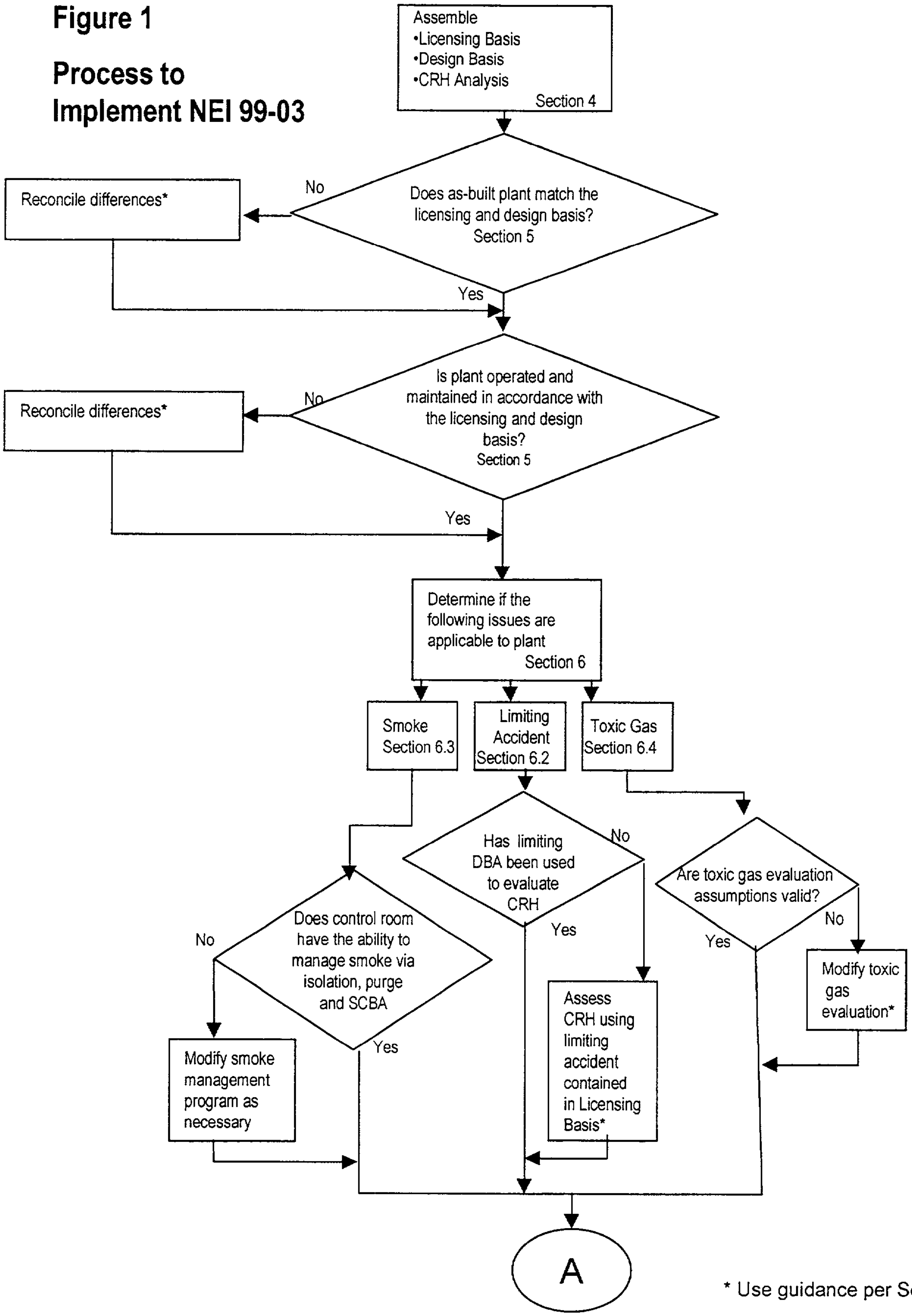
NEI 99-03 provides a comprehensive set of tools that function together to improve overall control room envelope (CRE) integrity program performance. Since plants differ in the degree to which their CRE configuration is being maintained, this document is designed to be used in total or in part to satisfy the needs of particular licensees.

The process described in NEI 99-03 is designed to ensure that the licensing and design bases associated with control room habitability (CRH) are verified and maintained. The document's guidance facilitates adequate protection of control room operators against the effects of postulated releases of radioactive particulates and gases, toxic gas or external smoke. Figure 1 illustrates the process that will demonstrate adequate protection of the control room operators based on the limits of the plant's existing design and licensing bases. The document also guides the development of a Control Room Envelope Integrity Program to facilitate long-term maintenance of the control room envelope. In conclusion, these elements define an acceptable method to demonstrate and maintain CRH.

NEI 99-03 does not provide guidance relating to the risk significance associated with maintaining CRH. Evaluation of the risk associated with potential operator exposure from radiological releases is not practical because probabilistic risk assessment (PRA) models assume that the operator will be able to stay on station during an accident and the core damage has occurred when CRH becomes a concern.

Figure 1

Process to Implement NEI 99-03



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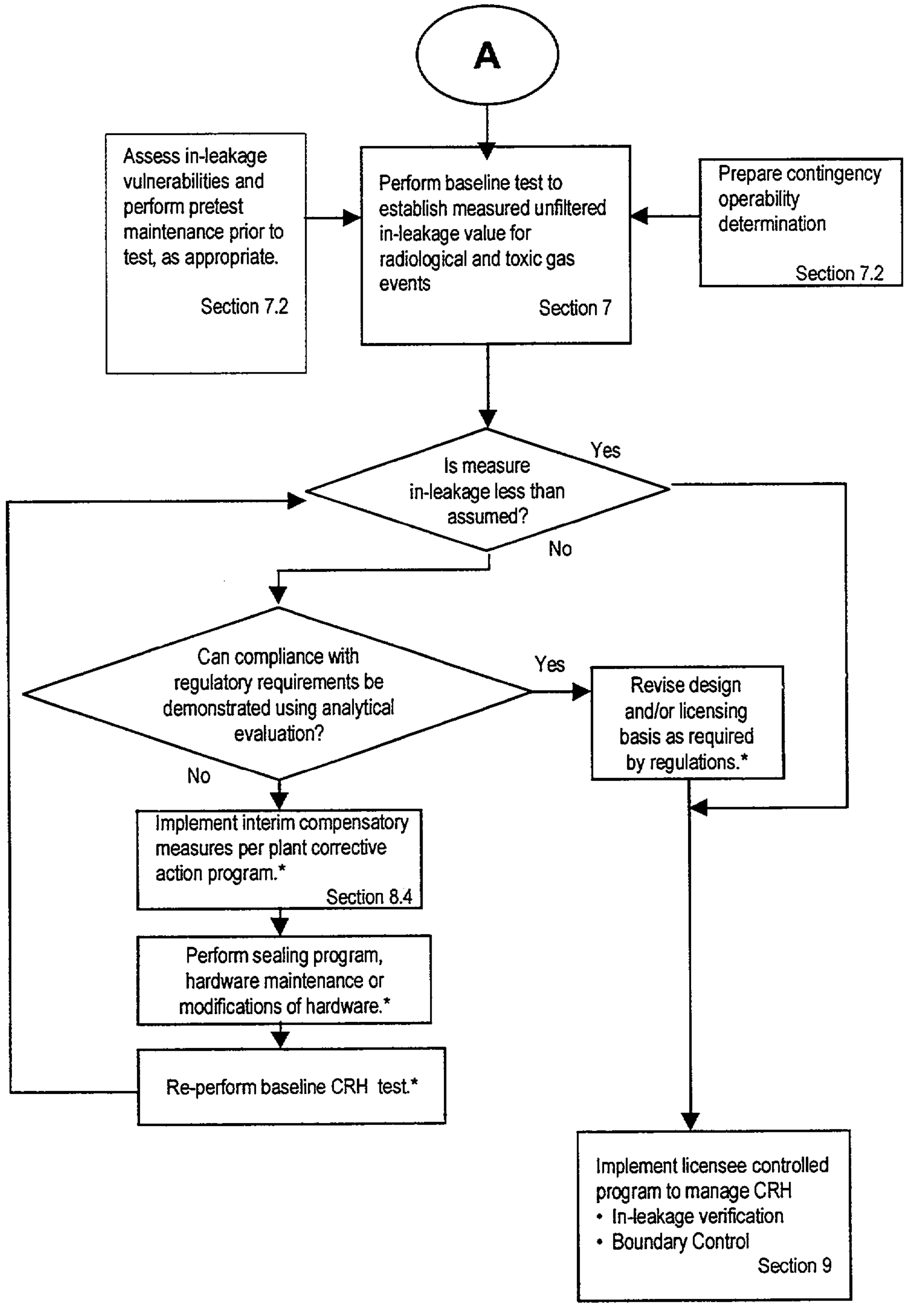

* Use guidance per Section 8

1.2 HISTORY

In 1971, 10CFR50, Appendix A, General Design Criterion 19 (GDC-19) became a regulation. GDC-19 requires protection of the control room operator under normal and accident conditions against the threat of radiological hazards. Following the Three Mile Island (TMI) accident in 1979, actions mandated by the Nuclear Regulatory Commission (NRC) required licensees to evaluate their control rooms to assure adequate protection of operators. Licensees have different levels of commitment to GDC-19 and the TMI action plan. The licensee's commitments are defined in the plant's licensing basis.

Since the mid-1980s, the NRC has communicated concerns about the adequacy of control room designs. In the mid-1990s, testing of some control room envelopes indicated that key assumptions supporting the radiological dose analysis might be non-conservative. In 1998, the NRC held a public workshop to address CRH concerns. In late 1998, the NRC and the industry agreed to work together on issues affecting CRH and develop this guidance document for resolving those issues.

Appendix A contains additional details on the history of the CRH issues and provides references to related documents.

1.3 DOCUMENT ORGANIZATION

This document defines a process for assessing a plant's CRH design and licensing bases to assure that they are established and maintained throughout the life of the plant. The main body of the document describes the general process. The appendices provide technical background information and detailed guidance for completing the assessment.

The main body of the document is divided into three parts:

- Background
- Assessment Process
- Establishment and Maintenance of CRE Integrity

Part 1, *Background*, provides basic information related to CRH, establishing a foundation from which the CRH assessment can proceed. It is composed of:

- Section 1, *Introduction*
- Section 2, *Regulatory Requirements and Guidance*
- Section 3, *Industry Issues Associated with Control Room Habitability.*

Part 2, *Assessment Process*, describes an approach to determine if the plant configuration and operation are consistent with the CRH licensing basis and analysis. It is composed of:

- Section 4, *Determining CRH Licensing Basis*
- Section 5, *Comparing Existing Plant Configuration and Operations With Licensing Bases for CRH*
- Section 6, *Assessing Industry Issue Applicability*
- Section 7, *Measuring Air In-leakage (Baseline Test)*
- Section 8, *Dispositioning and Managing Discrepancies.*

Section 4, *Determining CRH Licensing Basis,* provides guidance on identifying and assembling the current plant licensing and design bases. This information is to be compared with actual plant configuration and operation as described in Section 5, *Comparing Existing Plant Configuration and Operations With Licensing Bases for CRH.*

Section 6, *Assessing Industry Issue Applicability,* provides guidance for determining the plant specific applicability of the industry issues described in Section 3 and for identifying appropriate actions.

Section 7, *Measuring Air In-leakage (Baseline Test)*, recommends the completion of a baseline test to determine the amount of unfiltered CRE in-leakage and provides guidance on its performance. The test may be performed using the ASTM E741 tracer gas methodology or a component test methodology. In addition, guidance is provided defining alternate test methods. The purpose of this baseline test is to determine if the in-leakage is consistent with that used in the CRH evaluation.

Section 8, *Dispositioning and Managing Discrepancies,* discusses how to manage degraded and nonconforming conditions consistent with the licensee's Corrective Action Program. Sections 5, 6 and 7 refer to this section when degraded or nonconforming conditions are identified.

Part 3, *Establishment and Maintenance of CRE Integrity*, provides guidance on implementing a licensee controlled program to manage CRH after the evaluations of Part 2 are completed. It is composed of one chapter:

- Section 9, *Long-Term CRE Integrity Program*

The recommended program includes periodic assessment of in-leakage and maintenance of CRE integrity.

2 REGULATORY REQUIREMENTS AND GUIDANCE

2.1 PURPOSE AND SCOPE

This section identifies documents containing regulatory requirements and guidance related to CRH. It provides background information to assess which requirements are applicable to control room design, analyses and procedures. Appendix B provides additional details on the requirements and documents discussed in this section.

2.2 REGULATORY REQUIREMENT – GENERAL DESIGN CRITERION 19

The CRH requirement for operator radiological exposure is stated in 10CFR50 Appendix A, Criterion 19, *Control Room,* and is generally applicable to all utilities. However, not all plants are licensed to this requirement. Some plants only committed to selected aspects of GDC-19 or may have other similar commitments defining acceptable operator radiological exposure. The text of this rule has been included in Appendix B of this document.

GDC-19 provides acceptance criteria for radiological events. For most plants, the operator dose criterion limit is 5 rem whole body or its equivalent to any part of the body for any postulated design basis accident. NRC Standard Review Plan Section 6.4 interprets this requirement to be satisfied by a thyroid or a beta skin limit of 30 rem.

The Alternative Source Term (AST) Rule, 10CFR50.67, defines a different radiological limit of 5 rem Total Effective Dose Equivalent (TEDE) for licensees implementing the AST.

Acceptance criteria for non-radiological accidents (i.e., toxic gas release) are provided in other guidance documents discussed below.

2.3 REGULATORY GUIDANCE

An overview of various types of documents that supply regulatory guidance relative to CRH is discussed in this subsection. The documents discussed here are not requirements. Each plant must determine the extent to which its licensing basis includes commitments to each of these guidance documents. A plant may have committed to the guidance or taken exception to it, either in whole or in part.

2.3.1 Regulatory Guides

Regulatory Guides provide one acceptable approach for satisfying regulations. Licensees may propose alternative approaches for NRC approval. In this sense, use of a regulatory guide is voluntary. The approved approach becomes part of the plant's licensing basis and changes thereto are subject to 10CFR50.59. Appendix B lists several regulatory guides that relate to the evaluation of CRH systems and dose evaluations. The Regulatory Guides listed in Appendix B include guidance on the topics of accidents, analysis assumptions and system design.

2.3.2 NUREGs

NUREGs provide results of NRC research and general information on selected topics. NUREGs associated with CRH are identified in Appendix B.

NUREG-75/087, subsequently revised and issued as NUREG-0800, describes the manner in which the staff performed reviews of applications for an operating license or a construction permit. Sections of these NUREGs detailed the manner in which accident analysis consequences and control room habitability may have been assessed.

Action item III.D.3.4 of NUREG-0737, *Clarification of TMI Action Item Requirements,* deals directly with CRH. NRC Orders associated with this NUREG directed applicable plants to re-confirm compliance with GDC 19. The licensee response to the TMI action item may be a key consideration as a plant researches its licensing basis.

2.3.3 Information Notices

Information Notices (IN) previously called Inspection and Enforcement Notices , (IEN) are issued to inform licensees about events, issues and generic observations. The documents do not require a response, and plants generally do not have docketed commitments to incorporate changes into the design or operation practices as a result of INs or IENs.

IN/IENs applicable to CRH are identified in Appendix B. These documents provide information on plant designs or events with an impact on CRH.

2.4 GENERIC ISSUES

There have been two NUREG-0933 generic safety issues (GSI) related to CRH: B-66, *Control Room Infiltration Measurements*, and 83, *Control Room Habitability*. GSI B-66 was closed in 1983. GSI-83 was identified later that same year and is still open. These GSIs are summarized in Appendix A. This information provides insight into the historical evolution of the CRH concerns.

3 INDUSTRY ISSUES ASSOCIATED WITH CONTROL ROOM HABITABILITY

3.1 PURPOSE AND SCOPE

This section summarizes a number of industry issues associated with control room habitability. It is intended that licensees will review these issues for applicability to their plant and take the appropriate action to resolve any potential concerns.

At the July 1998 CRH workshop, the NRC discussed the following issues:

- Licensing basis different from as-built plant
- Analyses different from as-built or as-operated plant
- Design basis accident (DBA) analyzed not most limiting for CRH
- Smoke infiltration
- Toxic gas evaluation
- Control room in-leakage greater than assumed in analyses

This section discusses these issues. Sections 5, 6 and 7 provide guidance on assessing the applicability of each issue for a particular plant and dispositioning the results.

3.2 LICENSING BASIS DIFFERENT FROM AS-BUILT PLANT

During review of license amendments, licensees and the NRC staff have observed that some licensees have introduced inconsistencies between the plant's licensing basis and the as-built plant. Differences between the description of the control room envelope, the HVAC systems controlling the airflow within the envelope, and the as-built condition of the plant have been identified and documented. Modifications to systems or the envelope boundary may have inadvertently changed the CRH response. Also, maintenance or operations activities may have resulted in repositioned dampers that could influence the system response or associated control room boundary integrity. Section 5 provides guidance on assessing this issue.

3.3 ANALYSES DIFFERENT FROM AS-BUILT OR AS-OPERATED PLANT

The design analyses used to determine the operator exposure to a radiological event or a toxic gas event include several inputs that are based on system design parameters and assumed system operation. Licensees and the NRC have observed that some systems may have been operated differently from the assumptions or values used in the analyses.

Power up-rates, steam generator replacement and alternate repair criteria for steam generator tubing are examples of modifications that could impact the results of a licensee's CRH analysis. Licensees should assess the impact of these types of changes on CRH and the supporting analyses. Section 5 provides guidance on assessing this issue.

3.4 DBA ANALYZED NOT MOST LIMITING

Each plant is required to analyze the limiting design basis accident relating to CRH within the scope of its licensing basis. Most licensees and the NRC assumed that the large break LOCA was the limiting DBA for CRH. Reanalysis has shown that other licensing basis accidents can be more limiting. Section 6.2 provides guidance for assessing this issue.

3.4.1 Adjacent Unit Accident (a special case)

A few plants are within the exposure range for a DBA release from a nearby nuclear plant or have separate control rooms for multiple units on the same site. An accident in an adjacent unit should not prevent the safe shutdown of an operating unit.

3.5 SMOKE INFILTRATION

Smoke infiltration may be a CRH concern if there is significant in-leakage from outside the envelope. In this situation, smoke from external sources could challenge the ability of the operator to shut down the reactor from within the control room or remotely. No regulatory limit exists on the amount of smoke allowed in the control room. Therefore, the plant's ability to successfully manage smoke infiltration must be assessed qualitatively. Section 6.3 and Appendix E provide guidance for this assessment.

3.6 TOXIC GAS EVALUATION

Licensees have evaluated their susceptibility to toxic gas events, typically in accordance with Regulatory Guides 1.78 and 1.95. Those that are susceptible have committed to the NRC to take appropriate actions. The sources of toxic gas releases may change over time and should be assessed periodically. Section 6.4 and Appendix G provide guidance on toxic gas assessments.

3.7 CONTROL ROOM AIR IN-LEAKAGE GREATER THAN ASSUMED

Tracer gas tests have been conducted at approximately 25 percent of nuclear power plant control rooms to determine the total amount of air in-leakage (filtered and unfiltered). Test results show that measured in-leakage has been greater than the amount assumed in CRH design basis analyses. In some cases the difference was significant. This is a concern because the control room in-leakage value is an input to the evaluation of both radiological and toxic gas events. Section 7 provides guidance for assessing control room in-leakage.

3.7.1 Radiological Considerations

The unfiltered in-leakage rate is one of several inputs into the analyses used to determine operator doses. The term *unfiltered* refers to air leaking into the control room envelope that does not pass through an appropriate filtration device. With higher unfiltered in-leakage, the iodine removal credited in the accident analyses may be inaccurate and non-conservative. As a result, increased control room unfiltered in-leakage could result in the reactor operator being exposed to a larger dose than previously analyzed.

3.7.2 Toxic Gas Considerations

In-leakage is also a concern for toxic gas events. Increases in in-leakage may negate the conclusions of previous toxic gas analyses. The amount of in-leakage during a toxic gas event may not be the same as for a radiological event due to differences in plant alignment. A typical control room response to a radiological event is to isolate and pressurize; whereas a typical response to a toxic gas event is to isolate only. The plant alignment should be considered when determining the amount of in-leakage to be used in the toxic gas analysis.

PART 2 – ASSESSMENT PROCESS

4 DETERMINING CRH LICENSING BASIS

4.1 PURPOSE AND SCOPE

This section provides guidance that will help the licensee identify the plant's control room habitability licensing basis. This information will be used throughout the remainder of this guidance document.

4.2 UNDERSTANDING THE CONCEPT OF LICENSING BASIS

To apply the guidance provided by this document, the licensees must know the plant's CRH licensing basis. In addition, knowledge of the CRH licensing basis will assist licensees in:

- Making informed decisions regarding proposed changes to the physical plant or its operation,
- Responding to questions from the NRC staff when license amendments are proposed,
- Dispositioning corrective actions for degraded conditions and
- Processing changes that could affect the level of protection provided to control room operators.

There are several terms used in reference to basic information for systems, structures and components:

- Design basis
- Supporting design information and
- Licensing basis.

Understanding the difference implied by these terms is important to determine what is included in the licensing basis and what is not. A more detailed explanation of these concepts is provided in NEI 97-04, *Design Basis Program Guidelines*. The following paragraphs provide an overview of the concepts involved.

4.2.1 Design Basis

Design basis is defined in 10CFR50.2 as follows:

"Design bases means that information which identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally "accepted state-of-the-art" practices for achieving functional goals or (2) requirements derived from analysis (based on calculations and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals."

The design basis consists of both design basis functions and design basis values. Design bases functional requirements are derived primarily from the principal design criteria (e.g., GDC-19 of Appendix A to 10CFR50) and other NRC regulations, such as the Emergency Core Cooling Systems, Station Blackout and Anticipated Transient Without Scram rules.

4.2.2 Supporting Design Information

Supporting design information includes other design inputs (e.g., unfiltered in-leakage), design analyses and design output documents. Supporting design information may be contained in the UFSAR or other documents. Some supporting design information is submitted to the NRC and some is not. Supporting design information is controlled in accordance with Criterion III of 10CFR50 Appendix B.

4.2.3 Licensing Basis

The *licensing basis* for a plant establishes its compliance with regulatory requirements. It describes how the plant meets the appropriate regulations and may also include, for example, exceptions to specific regulatory guidance that were approved by the NRC in a Safety Evaluation Report.

Figure 2 presents the relationship of the design basis to the license basis. The design basis is a subset of a plant's licensing basis. It is important for licensees to establish the scope of regulatory requirements to which they are licensed. In general, a licensee is committed to regulations in place at the time of plant licensing and to other criteria contained in commitments made since the original license was granted.

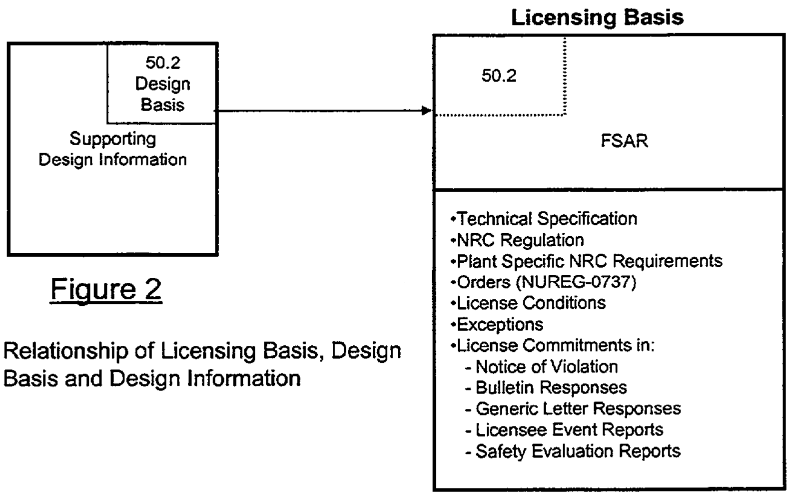

Figure 2

Relationship of Licensing Basis, Design Basis and Design Information

One important source of licensing basis information is licensing actions initiated after a plant's operating license is granted. A Safety Evaluation Report written by the NRC typically identifies approved licensing actions. The NRC SERs typically document the boundaries of the licensing action proposed by the applicant, the applicant's analysis, the staff's evaluation of the proposed action and the basis of the staff acceptance. Generally, the information in an SER can be considered as part of the licensing basis to the extent that the SER reflects the information docketed by the licensee or documents the basis for the NRC acceptance.

In summary, a plant's licensing basis consists of only those items that it is required by regulation to meet or to which it has formally committed itself in written correspondence signed by a utility representative with the appropriate authority. NEI 99-04, *Guidelines for Managing NRC Commitment Changes*, provides more information on this subject.

4.3 LICENSING BASIS SOURCES

Licensees document compliance with regulatory requirements in various documents. Licensees should consider the following documents when performing a design basis review:

- UFSAR
- Licensing correspondence that contains commitments and exceptions to applicable regulatory requirements and regulatory guidance
- Operating license and amendments
- Technical Specifications and their bases

- Plant-specific NRC staff requirements and positions applicable to the plant, whether originating from 10CFR50, SERs, generic communications or regulatory guides
- Other plant specific licensing documents that list licensing parameters, values and assumptions.

Appendix B provides a description of the regulatory requirements and guidance related to CRH.

4.4 PERFORMING THE LICENSING BASIS REVIEW

NEI 97-04, *Design Basis Program Guidelines,* provides guidelines for identifying design basis information. Even though design basis information is only a subset of the licensing basis, the process identified in NEI 97-04 is useful for assembling the plant's licensing basis.

4.5 ASSEMBLING THE CRH ANALYSIS

An important part of a control room design basis is the CRH analysis. This analysis is typically performed during initial plant design to determine operator exposure to the hazards produced by DBAs. For most plants, a CRH analysis will not be available as a stand-alone document. Rather, the licensee will need to assemble it from its component parts. These parts should be found as written design basis documentation and licensing commitments. The following types of information should be accessed to assemble the CRH analyses:

- Design basis accident analyses within the plant's licensing basis. Licensees should have a thorough understanding of the design basis accidents analyzed for CRH and should know the analysis results (such as radiological consequences) to ensure that the most limiting accident is identified.
- Specific performance requirements for components that provide a radiological, toxic gas or smoke mitigation function along with component performance data.
- Analysis inputs, such as the amount of unfiltered in-leakage or control room volume, their bases and source documents. For example, inputs such as occupancy factors may have been adopted from the Standard Review Plan.
- All modes of control room ventilation system operation and system alignments necessary to mitigate radiological and toxic gas events and external fires.
- Component functions. The design basis documents for controlling the performance of these components should be identified and reviewed to ensure consistency. Such documents include:
    - Design specifications
    - P&IDs

  - Logic diagrams
  - Wiring diagrams
  - Performance test acceptance criteria.
- Technical Specification performance limits and surveillance requirements for credited components.
- Commitments regarding operation of the control room envelope and other requirements regarding operation of the control room envelope. These may be identified in such documents as the licensee's Updated Final Safety Analysis Report (UFSAR), Design Criteria Memoranda, operating procedures, surveillance test procedures, etc.
- License submittals that may have an effect on CRH such as steam generator replacement, steam generator alternate repair criteria and power uprates.

4.6 DOCUMENTATION OF THE EXISTING PLANT CRH LICENSING AND DESIGN BASIS

The identification of the CRH licensing basis must proceed methodically and be carefully documented. The process should ensure that all source documentation is reviewed. When licensing basis information is identified, it should be captured and accurately referenced to allow subsequent retrieval in its original context to facilitate review and verification if necessary.

The implementation of a CRH licensing basis identification program will identify open items that may include questions, concerns and cases of missing information. Open items that do not conform to the licensing basis or have potential safety significance are considered discrepancies. The CRH licensing basis identification program must include means to identify, capture and disposition these items. Guidance on managing discrepancies is provided in Section 8.

A process for documenting the information should be developed that allows its efficient use by subsequent steps of this guidance document.

5 COMPARING EXISTING PLANT CONFIGURATION AND OPERATIONS WITH LICENSING BASES FOR CRH

5.1 PURPOSE AND SCOPE

After the control room habitability design and licensing bases have been compiled, a comparison to the control room (CR) system configuration, operation and maintenance should be performed.

This comparison is needed because after years of operation, new procedures and methods of operation, maintenance and testing may have been developed and revised during the years of plant operation. Systems may be operated differently from the assumptions or values used in analyses that determined operator exposure from radiological or toxic gas events. Given this change process, it is prudent to confirm that current practices are consistent with the licensing basis.

This section provides a guide for performing this assessment.

5.2 REVIEW THE AS BUILT CONTROL ROOM ENVELOPE AND CONTROL ROOM VENTILATION SYSTEMS

Review the as built configuration to ensure that the construction and configuration satisfy the design and licensing bases. For example:

- Plant drawings should be reviewed to ensure that the design would provide the desired CR isolation function and support the in-leakage assumptions. For example, confirm that assumed automatic response functions have been implemented.
- Component specifications should be reviewed to ensure that the licensing and design bases are consistent with current design. For example:
  - Do fans provide the required flow rates?
  - Do dampers provide the design leak tightness?
  - Are duct design requirements for construction quality consistent with leakage assumptions?
- A system walk down should be performed to ensure that the actual field configuration agrees with the plant drawings/design. For example:
  - Are the air sources from the appropriate location?
- The control room envelope assumed for in-leakage evaluations should be compared to that identified in plant documents or surveillance procedures to ensure the identified boundaries are consistent.

5.3 Review the Normal and Emergency Operating Procedures Affecting the Control Room Ventilation Systems

Review the plant operating procedures to ensure that the licensing and design bases are maintained. This review should include procedures for both normal and off-normal conditions. For example:

- Ensure that emergency operating procedures (EOPs) do not invalidate the licensing basis while attempting to restore room cooling in certain situations.
- Ensure that normal operating procedures align the system to establish the proper flow paths. Damper settings should be correct to establish the necessary flow rates and isolation capability.
- Ensure the EOPs place the control room ventilation system in the correct configuration for the existing plant condition. For example, the proper configuration may be recirculation for a toxic gas event or pressurization for a radiological release.

5.4 Review the Testing Procedures Affecting Control Room Ventilation Systems and the Associated Envelope

Review testing procedures to assure consistency with the following:

- The procedures should adequately demonstrate the operability of the intended components.
- The procedures should ensure that the envelope is not inadvertently breached, or otherwise made inoperable during the test.
- The system should be properly realigned after completion of the test.
- Post-maintenance testing should be sufficient to ensure that the system is functional and properly configured before being returned to an operable state.
- If components are being tested for in-leakage, the test configuration and test conditions must appropriately reflect the leakage expected under accident conditions.

5.5 Review the Maintenance Practices and Procedures for Effect on CRH Requirements

Assess maintenance procedures to assure that they do not adversely impact the control room envelope integrity or render a system inoperable. For example:

- Maintenance planning should consider the required operability of control room ventilation components for the current plant-operating mode, as defined in Technical Specifications.
- Maintenance practices affecting structures should be reviewed to ensure that the CR envelope could not be inadvertently breached.

- Maintenance procedures for system components should address CR integrity requirements. Procedures should note that removal of inspection plates or opening access doors might constitute a breach of the CR envelope.
- Breach control programs and procedures designed to seal, maintain and inspect the integrity of the control room envelope should be of sufficient detail to examine all likely sources of control room in-leakage. Easily damaged components, such as door seals, should be considered for increased scrutiny.

5.6 REVIEW THE PLANT MODIFICATION PROCEDURES FOR CONSIDERATION OF THE CRH REQUIREMENTS

Evaluate the design control procedures to ensure that changes that may have a direct or indirect impact on CRH are properly evaluated. Design change procedures should include a check of the effect of the modification on the control room envelope integrity. For example:

- Direct modification of the ventilation system could have the effect of changing the system's performance characteristics.
- Modification of ventilation systems in areas adjacent to the control room could affect the in-leakage values.
- Electrical work such as installing new conduit or pulling cable could create new in-leakage paths.
- Installing floor or equipment drains could result in unexpected in-leakage paths.

5.7 REVIEW THE CRH ANALYSES

Review the CRH analyses to assure that they are consistent with the licensing basis and with the current control room envelope and HVAC procedures and configuration. For example:

- Do the system lineups assumed in the CRH analyses agree with the current procedures?
- Are the in-leakage assumptions in the CRH analyses (radiological and toxic gas) valid?
- Are the assumptions in the CRH analyses reasonable in light of current operations and configurations?

5.8 IDENTIFIED INCONSISTENCIES

Any inconsistencies between the existing plant CRH configuration and operations and the licensing bases should be reconciled per the plant's corrective action program as described in Section 8.

6 ASSESSING INDUSTRY ISSUE APPLICABILITY

6.1 Purpose and Scope

This section provides guidance for evaluating the plant specific applicability of the following industry issues introduced in Section 3.

- Limiting design basis accident
- Smoke infiltration
- Toxic gas evaluation

This section also recommends actions to address the applicable industry issues.

6.2 Limiting DBA

The Large Break Loss of Coolant Accident (LBLOCA) DBA was frequently assumed by licensees to be the bounding accident for control room habitability. Licensees frequently used the LBLOCA dose analyses response to assess the adequacy of the CRH design. The impact of different plant configurations, responses or atmospheric dispersion from other accidents, including accidents at adjacent units, on the radiological consequences to the reactor operators may not have been adequately considered. In addition, changes to plant design or operations may not have been analyzed adequately over the spectrum of the plant licensing basis events for the effect on control room habitability.

6.2.1 Recommended Actions to Evaluate Limiting DBA

Examine each DBA listed in the UFSAR for which off site doses have been reported to determine the event that is limiting with respect to control room dose. UFSAR-described accident scenarios and postulated source terms should be combined with a control room model using appropriate system parameters and responses and event-specific atmospheric dispersion factors. If the facility license or UFSAR requires consideration of releases from accidents at an adjacent unit, these events should also be evaluated. If a new CRH limiting DBA is identified, corrective action in accordance with Section 8 and the plant's corrective action program should be taken. Appendices C and D provide additional guidance for performing these evaluations.

Factors that may influence the limiting accident with respect to CRH include:

- The impact of the control room isolation delay on radiological consequences to the reactor operators.

  - For accidents where the CRH features are actuated by containment isolation or safety injection (SI) signals[1], there is little or no delay. Where the CRH features are actuated by radiation monitor alarm signals, there may be a time delay to achieve control room isolation. In such cases, contaminated air may enter the control room for a longer period.
  - Radiation monitor configuration may affect the ability to actuate the CRH features in a timely manner.
  - Differences in source terms for the different postulated (and potential) accidents can have a significant impact on monitor response.
- Radiological release locations can dictate which analyzed accident is limiting. Some considerations are:
  - The distance between the control room intake and release points may be different for each postulated accident.
  - Release points for some accidents may be in a direction frequently downwind of the control room intake, while those for other accidents may usually be upwind.
  - A ground-level release associated with a non-LOCA event may be more limiting than the elevated release associated with a LOCA at units with a secondary containment or enclosure building.
- Approved alternate repair criteria (ARC) for steam generators.
  - The main steam line break accident is generally the limiting accident with regard to control room habitability for plants with ARCs since such facilities may have maximized the postulated control room operator dose in order to maximize the number of tubes to which the alternate repair criteria can be applied.

6.2.2 Adjacent Unit Accidents

A special case of limiting DBA could be the presence of an accident release from an adjacent unit. The release point, atmospheric dispersion and postulated source term for the adjacent unit should be reviewed to assess the impact on the operating unit. This potential limiting DBA need only be considered if it is within the licensing basis of the plant evaluating its control room.

If there are adjacent units with separate control rooms, then an accident in one unit should not prevent the safe shutdown of the adjacent unit. Atmospheric transport mechanisms between the accident unit and the HVAC intakes to the operating unit control room should be reviewed for impact on CRH.

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1 Typically, control room isolation is activated by engineered safety feature signals such as containment high pressure or safety injection, or radiation monitors, or both.

6.3 SMOKE INFILTRATION

Recent control room in-leakage tests have identified that control rooms may be operated with significantly more in-leakage than previously assumed and therefore control room operators may be exposed to a greater amount of smoke infiltration in the event of an external fire. The increased smoke could make the control room uninhabitable and impair the operator's access to the remote shutdown locations. Licensees should consider if they are appropriately prepared to mitigate such smoke infiltration.

Currently, no NRC regulations exist to establish smoke concentration limits. Varied sources of fire produce different amounts of smoke with different compositions. 10CFR50 Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979, does not provide guidance in this area. Generic Safety Issue 148, *Smoke Control and Manual Fire-Fighting Effectiveness*, of NUREG-0933, stipulates that fires are anticipated to be of short duration relative to design basis accidents.

6.3.1 Recommended Licensee Action to Address Smoke Infiltration

Recognizing the importance of the smoke concern, it is recommended that licensees perform a qualitative evaluation of their smoke management capabilities. This guidance is provided in Appendix E. The licensee should assure that it could safely shutdown the reactor in the event of smoke infiltration into the control room.

6.4 TOXIC GAS EVALUATION

Control rooms are typically evaluated to assure that they can manage a toxic gas event. This evaluation may not have been performed since the early 1980s in response to Three Mile Island (TMI) (NUREG-0737) item III.D.3.4. The amount of in-leakage may be greater than that assumed in the existing evaluation. This concern is addressed in Section 7 and Appendix G. In addition, the sources of toxic gas may have changed over time and the existing evaluation may not account for the current toxic gas threats near the plant.

6.4.1 Recommended Licensee Action to Address Toxic Gas Evaluation

Appendix G provides guidance on toxic gas evaluations, and Section 9.3.3 provides recommendations on the frequency of the assessment. If a new toxic gas survey is appropriate, conduct an inventory of mobile and stationary sources of hazardous chemicals in the vicinity of the plant in accordance with Appendix G. If significant new toxic gas sources are identified, revise the control room toxic gas evaluation in accordance with Appendix G.

7 MEASURING AIR IN-LEAKAGE (BASELINE TEST)

7.1 PURPOSE AND SCOPE

As discussed in Section 3.7, unfiltered air in-leakage is one of numerous assumptions used in radiological and toxic gas evaluations. Approximately 25 percent of the control rooms have been tested recently for in-leakage. Each test result indicated that the actual measured value exceeded the value originally assumed in the accident analyses. As a result, this document recommends that a baseline test be performed to determine a numerical value for control room in-leakage that can be compared to the accident analyses assumptions and used to assess the integrity of the control room envelope. This section provides an overview of preparation for baseline testing, test performance and resolution of identified issues.

7.2 PREPARATION FOR TESTING

Prior to performing a baseline test, it is recommended that a system assessment be performed per the guidance provided in Appendix H. The system assessment includes a walkdown to identify (1) discrepancies in the envelope, and (2) components vulnerable to in-leakage. The system assessment should help to find potential in-leakage paths that are candidates for pre-test maintenance or design modifications. The licensee may choose to perform maintenance to eliminate any suspected in-leakage paths before performing the baseline test for in-leakage.

It is advisable to plan compensatory measures should the measured in-leakage exceed design basis assumptions. In addition, if the assumed design basis in-leakage allowance is small, consideration should be given to performing a reanalysis that provides a larger allowable in-leakage value. Experience has shown that small values of CR in-leakage may not be measurable due to the uncertainties in the test methods.

7.3 TEST PERFORMANCE

Licensees should perform a baseline test to determine the actual value of control room in-leakage for use in control room habitability analyses. The baseline integrated in-leakage test may be performed using a tracer gas test, a component test or an alternative equivalent methodology. The tracer gas test method is appropriate for all control room designs; however the component test should only be performed on control rooms designed to be maintained at a positive pressure. The component test method verifies that the control room pressure is greater than adjacent spaces and tests all components that cannot be shown to have a positive differential pressure relative to non-control room envelope areas. Appendix I further defines test methods. Note that blower door testing (ASTM E779) and smoke pencil testing (ASTM E1186) may be useful for troubleshooting localized or component in-leakage problems, but they are not recommended as an integrated test method.

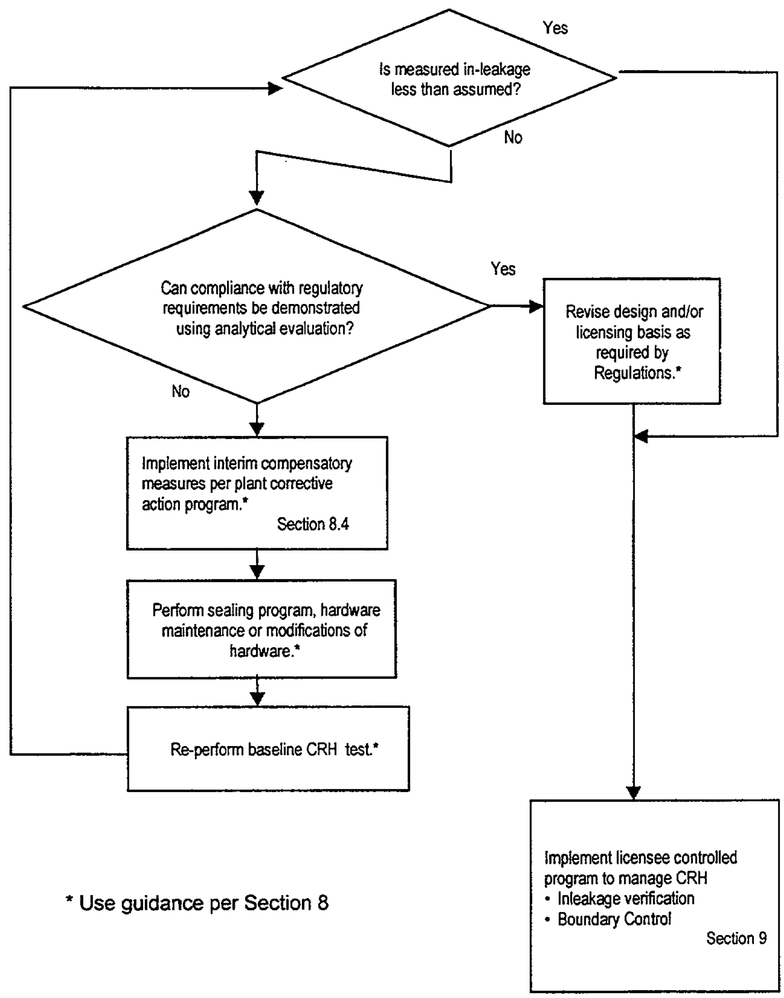
7.4 RESOLUTION OF IDENTIFIED ISSUES

Once a measured baseline unfiltered in-leakage value is determined, it should be compared to the value used in the CRH radiological and toxic gas analyses. If the measured in-leakage value is greater than the analysis input, the licensee should take corrective actions through its corrective action program discussed further in Section 8.

In addition, the measured in-leakage value should be evaluated from the perspective of smoke infiltration into the control room per Appendix E. If a qualitative evaluation indicates a concern, actions should be taken to address the condition.

Corrective actions may include reanalysis, a design change, sealing and re-baseline testing to ensure the design and licensing basis are met. Appendices C, D and G provide guidance that may be appropriate if reanalysis is performed. In addition, the alternative source term rule, 10 CFR 50.67, and Regulatory Guide 1.183 may provide additional relief when used for a reanalysis of the radiological challenges. Figure 3 reproduces the relevant portions of Figure 1 and demonstrates a process to follow.

Figure 3: Relevant Portions of Figure 1



* Use guidance per Section 8

8 DISPOSITIONING AND MANAGING DISCREPANCIES

8.1 PURPOSE AND SCOPE

Conditions adverse to quality must be promptly identified and corrected in accordance with 10CFR50, Appendix B, Criterion XVI. This is accomplished by each licensee's Corrective Action Program. The primary guidance for identifying and resolving degraded and nonconforming conditions is provided by Generic Letter (GL) 91-18, Revision 1, *Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Nonconforming Conditions*. Reportability criteria are specified by 10CFR50.72, *Immediate notification requirements for operating nuclear power reactors* and 10CFR50.73, *Licensee event reporting system.*

In addition, if changes are required, the criteria of 10CFR50.59, *Changes, tests and experiments*, may apply.

This section provides supplemental guidance for the evaluation of CRH discrepancies identified in Sections 5, 6 and 7. This section is a summation of practices already defined by the regulatory process and in place at operating plants.

8.2 GENERIC LETTER 91-18

GL 91-18 informed licensees of the issuance of a revised section to Part 9900, *Technical Guidance of the NRC Inspection Manual.* The revised section was entitled *Resolution of Degraded and Nonconforming Conditions.* This revised section provides guidance to NRC inspectors and provides explicit insights on appropriate actions to take when a degraded or nonconforming condition exists. The document directs assessment of the following:

- Operability determination
- Justification for continued operation
- Reasonable assurance of safety
- Compensatory measures (if used).

GL 91-18 describes three potential scenarios for addressing degraded and nonconforming conditions:

1. The licensee may restore the structure, system, or component (SSC) to the condition that is described in the licensing basis. For example, if the assumed control room in-leakage is explicitly described in the UFSAR and an in-leakage test reveals excessive in-leakage, the licensee may take corrective action to repair various seals and openings to reduce the in-leakage to within the UFSAR limit. See Appendix J.

2. The licensee may accept a condition "as-is" which results in something different from that described in the UFSAR or may modify the plant to something different than that described in the UFSAR. These options would be considered a change and would be subject to 10CFR50.59 unless another regulation applies. An example of this is modifying the control room envelope to enhance the leakage characteristics of the system. Another example would be revising the appropriate accident analyses to demonstrate the acceptability of increased in-leakage.

3. The licensee may take interim compensatory measures until the permanent corrective actions identified in items 1 and 2 above can be fully implemented. These compensatory measures may be subject to 10CFR50.59. For example, potassium iodide (KI) tablets and/or self contained breathing apparatus may be utilized to minimize operator dose until other actions are taken.

Section 8.4 describes these methods in more detail.

8.3 Determining Operability and Reportability

If a degraded or nonconforming condition is identified, appropriate action must be taken to maintain the plant in a safe condition. GL 91-18 provides detailed guidance with respect to performing operability determinations. As pointed out in Appendix I, it is advisable that contingency plans and operability determination actions be completed before in-leakage tests are performed. Such planning can provide insights about the baseline testing acceptance criteria. A licensee may want to determine the maximum in-leakage that can be accommodated:

- Within the current regulatory limits
- Using the current source term, but with the analysis improvements of Appendices C and D
- Using the alternative source term (10CFR50.67 and Regulatory Guide 1.183), with or without the atmospheric dispersion improvements of Appendix D.

In addition, the compensatory measures of Appendix F (or other, plant-specific compensatory measures) should be considered for use in case the above levels cannot be met.

The reportability evaluation ensures timely NRC notification of significant conditions or events relative to regulatory compliance. The corrective action process should ensure that an identified discrepancy is evaluated for potential reportability to NRC under the requirements of 10CFR50.72 and 10CFR50.73.

The basis for operability and reportability, including evaluations and analyses, should be documented and retained for future use.

8.4 METHODS AVAILABLE TO ADDRESS DEGRADED OR NONCONFORMING CONDITIONS

8.4.1 Compensatory Measures

Compensatory measures may be implemented in the short term to mitigate an identified discrepancy that may result in the plant being in an unanalyzed condition or outside its design or licensing basis (i.e., degraded or nonconforming condition per Generic Letter 91-18). Compensatory measures must provide a reasonable assurance of safety until final corrective actions are complete. Compensatory measures can consist of additional administrative or procedural controls, additional testing or inspection of system components, and additional protection provided to control room operators through the availability of self-contained breathing apparatus and/or potassium iodide tablets. Licensees must ensure that compensatory actions can be implemented under 10CFR50.59 or request prior NRC approval. Guidance regarding compensatory measures related to CRH is provided in Appendix F.

8.4.2 Dose Analysis Revision Option

A revised dose analysis may be part of the short-term justification for continued operation or part of the long-term resolution of the nonconforming condition.

Revision of the analysis of record for the dose consequences to the control room operator may be an acceptable method for addressing a condition different from that described in the UFSAR and for meeting the requirements of GDC 19. Revision of the dose analysis of record may be desirable in combination with plant modifications to improve the margin to regulatory limits. Appendices C and D provide acceptable methods that licensees may use to revise their dose analysis. Appendix C focuses on improvements in the existing approaches (e.g., based on 10CFR100, TID-14844 and Regulatory Guide 1.3 and 1.4) to accident dose analysis.

An option for consideration in the development of the final resolution of the degraded condition is to revise the licensing basis. An example of a new licensing basis would be the implementation of the Alternative Source Term based on 10 CFR 50.67 and Regulatory Guide 1.183.

An increase in previously calculated operator doses may require NRC review and approval. In addition, some changes to the licensing basis (e.g., AST) or analysis methodology may also require prior NRC approval in accordance with 10CFR50.59. Regulatory Guide 1.187 and NEI 96-07, Guidelines for 10CFR50.59 Safety Evaluations, provide detailed industry guidance to address criteria for making this determination.

8.4.3 Repairing or Modifying the Plant

The identified in-leakage source may be corrected by a repair of the physical condition or by sealing the leak path.

In some instances, a plant modification may be desirable. Licensees may decide to modify their control room envelope boundary by:

- Moving HVAC equipment within the CRE
- Replacing ducts with seam-welded heavy construction material to eliminate ducting as a leakage source
- Modifying system controls to change actuation signal timing
- Securing non-emergency ventilation systems that contribute to in-leakage during operation and pressurization
- Modifying the system modes of operation.

Modifications may require a retest to ensure that they were successful in elimination of the in-leakage and provide appropriate validation of the assumed new in-leakage value.

8.4.4 Technical Specification Changes

Degraded or nonconforming conditions may be addressed by technical specification changes. The nonconforming degraded condition may be eliminated if one of the parameters associated with the limiting accident is in the technical specifications and can be changed. Examples might be reactor coolant activity levels, containment leak rate and primary-to-secondary leak rates.

PART 3 - ESTABLISHMENT AND MAINTENANCE OF CRE INTEGRITY

9 LONG-TERM CRE INTEGRITY PROGRAM

9.1 PURPOSE AND SCOPE

This section defines a control room envelope integrity program to verify control room habitability during the operating life of the plant.

9.2 CRE INTEGRITY PROGRAM

Many activities over the life of the plant may challenge CRH. Physical design modifications may affect the envelope or the control room HVAC systems. Changes in maintenance or operating practices may influence material condition or affect the assumptions in the licensing basis. New sources of toxic gas external to the plant site or design changes may affect inputs and assumptions in the plant control room habitability analyses. Thus, it is essential that CRE integrity be considered over the life of the plant.

Attributes of a CRE integrity program include:

- Periodic evaluations
  - System material condition
  - Post-maintenance activities
  - In-leakage assessments
  - Toxic gas evaluation
- Configuration control (design and operation)
  - CRE boundary / breach control
  - Procedure control
  - Toxic chemical control
  - Design change control
  - Safety analyses control
- Training
- Testing.

These attributes are discussed in the following sections.

9.3 PERIODIC EVALUATIONS

The baseline assessment and test program (per Sections 5, 6 and 7) will demonstrate that the CR design, configuration, performance and operation meet the plant design and licensing bases. However, control room envelope integrity is subject to change over time. A periodic assessment of the systems, components and key analysis assumptions is recommended to ensure optimal operation and early detection of problems and that the licensing basis is being maintained. The periodic assessment may involve design reviews, material condition assessments, inspections and/or testing as described in the following subparagraphs.

9.3.1 System Material Condition

Testing of some mechanical CR HVAC components, such as emergency supply fans and charcoal filters, is normally required by the plant Technical Specifications. Such testing may identify degradation of these components. However, degradation of various other components whose impact on CRH may be less obvious (e.g., manual dampers, control loops, non-safety related ventilation in adjacent spaces) can also affect safety analysis assumptions of CRE integrity. In addition, passive components such as door seals, penetration seals and wall/floor joints can degrade over time, affecting CRE integrity.

The material condition of the above components should be periodically assessed to ensure there is no degradation that would jeopardize integrity assumptions. In addition, the performance of HVAC systems in areas adjacent to the CRE should be reviewed to ensure that the licensing basis is maintained.

9.3.2 Post-Maintenance Activities

During the time interval between periodic assessments and/or testing, various maintenance activities will occur that affects either the control room envelope or the performance of the control room HVAC system. This may result from preventive maintenance, corrective maintenance or implementation of modifications. It is important to perform a proper post-maintenance test (PMT) following these activities to ensure that the integrity of the CRE is maintained. The actual PMT may be a simple inspection to ensure that a gasketed surface has been sufficiently tightened to eliminate air gaps or it may be a full in-leakage test if a major modification has significantly changed the boundary of the CRE.

The following examples are provided to illustrate possible PMTs that may be used to ensure that CRE integrity is maintained:

- A PMT that is performed under guidance of other documents, such as ANSI-N510 for filter changeout, would not require additional testing in accordance with this document.
- A pipe that penetrates the CRE has a flange mounted pressure transmitter that requires replacing. The flange has a bolted gasket connection that is fully accessible for inspection. An adequate PMT could be a visual inspection to ensure that proper gasket crush is achieved after the new transmitter is installed.

- A door seal requires replacing. The geometry of the gap between the door and the frame is such that a visual inspection is difficult to perform. An adequate PMT could be the use of a "smoke pencil" to verify that the door gasket has been properly installed to minimize leakage.
- A major modification has been performed to incorporate the CR HVAC equipment room into the CRE. A full in-leakage test may be required to ensure that the new configuration still meets the in-leakage assumptions used in the accident analyses.
- A modification has been performed on systems, structures and components outside the CRE that may affect CRE integrity. The complexity of the PMT would depend upon the effect of the modification on CRE integrity.

9.3.3 In-Leakage Assessments

The design configuration and material condition of the CRE must be periodically assessed to ensure that the CRE in-leakage assumption remains valid. This assessment may be only an engineering evaluation; however the evaluation may determine that a need to retest CRE in-leakage exists. It is recommended that the first reassessment be performed within one to three years of the baseline test. Subsequent to the first reassessment, the frequency of subsequent reassessments of the CRE is dependent on plant specific considerations as discussed below. The time of the next assessment should be defined at the conclusion of each assessment based on the findings.

All assessments should be documented and retrievable.

9.3.3.1 Assessment Scope

The objective of the periodic assessment is to ensure that CRE in-leakage assumptions remain valid. The scope of the assessment will vary between plants and over time depending on the condition of the CRE and CRE Integrity Program. For example:

- If there is an effective CRE Integrity Program in place and minimal modification to the CRE has occurred, the periodic assessment may consist of an engineering evaluation of the CRE condition that takes into account modifications and age related degradation.
- If the potential for challenges to CRH assumptions are more probable, the assessment may also include limited testing on components that have been modified or that are vulnerable to unfiltered in-leakage or age related degradation.
- In the case of a limited CRH Program or extensive changes to the CRE boundary, a more extensive assessment should be considered including performance of an integrated in-leakage test.

Appropriate CRH Program personnel should determine the proper scope for each in-leakage assessment. The basis for the scope of each assessment should be documented and retained with CRH Program records.

9.3.3.2 Assessment Frequency

The established assessment period must ensure the integrity of the CRE. It should be based on the safety significance and historical performance of the CRE and the available in-leakage margin. The CRH program should contain performance criterion for each of the factors contained in this section. Assessment frequency should consider:

- in-leakage rate limits
- conditions that are indicative of or affect performance
- performance evaluations of system components (see also paragraph 9.3.1)
- comparison to previous assessments to examine the performance history of the overall CRE to limit in-leakage.

The assessment should demonstrate that the in-leakage rate is less than the assumed in-leakage rate with margin as specified by the plant.

The following factors should be considered when establishing a frequency for periodic reassessment and the need to retest as opposed to performing an engineering evaluation. Typical frequencies could be based on refueling outage intervals, (An 18-month interval is used in examples below. Intervals of 5 years or 10 years may be used depending on margins and the following factors.) This is not an all-inclusive list. The basis for an acceptable margin is plant-specific. Each licensee is to assess its own situation.

- Confirm that control room envelope breaching controls are in place (Appendix K). Without breaching controls in place there can be no assurance that the boundary integrity will remain intact due to maintenance and/or modification activities. The adequacy of the breaching controls can be evaluated by determining the number of breaching problems that occurred or were prevented.
- Identify the in-leakage vulnerabilities. If the number of vulnerabilities or the magnitude of the potential leakage is large, then a more frequent assessment may be needed. Vulnerabilities should not be counted individually if they can be associated with a single component.
- Determine the available differential pressure margin between the CRE and adjacent spaces. If the measured differential pressure is smaller, more frequent assessments may be necessary to ensure that differential pressure is maintained. A suggested margin would be 0.05 inches wg above the design basis requirement. The suggested margin of 0.05 inches water guage (wg) is based on engineering judgment and experience. It is a high enough value to allow accurate measurement while providing reasonable

assurance that flow will be from the area of higher pressure to lower pressure. It is also sufficiently greater than the typical TS value (1/8 inches wg) to allow correction without going below the TS limit.

- The typical differential pressure maintained by pressurized control rooms is 1/8 inches wg. Therefore, a total pressure differential of 0.175 inches wg to outside atmosphere is recommended.

- For licensees that do not have limits for pressure differences between internal building spaces, 0.05 inches wg is recommended. The suggested margin of 0.05 inches wg is based on engineering judgment and experience. It is a high enough value to allow accurate measurement while providing reasonable assurance that flow will be from the area of higher pressure to lower pressure without the air flow into the adjacent space being overly burdensome to the utility.

- Compare the baseline test measured in-leakage to the design basis analysis in-leakage assumption for both radiation and toxic gas considerations. A small margin may require a more frequent assessment. This margin would be plant specific, but an example of suggested margin would be either 10 percent of the acceptable limit or the testing uncertainty value if it is larger.
- Confirm that maintenance practices are in place to assure that the boundary is maintained. This includes periodic inspections and preventative maintenance (Appendix J). Verify that periodic maintenance is being performed satisfactorily within the required frequencies.
- Confirm that the plant configuration control program evaluates the effect of modifications on CRE integrity. Determine the number and complexity of design changes that may have affected CRH. Numerous or complex modifications may indicate the need for more frequent assessment.
- Examine industry operating experience.
- Assess plant material condition. Degrading material condition may indicate the need for more frequent assessment.

As an example, CRE integrity that meets the following attributes may justify an assessment frequency of 10 years:

- An effective CRE breach control program is in place.
- The CRE has a small number of vulnerabilities to in-leakage.
- The measured CRE boundary differential pressure is significantly greater than 1/8 inches wg.
- The measured CRE in-leakage is considerably less than the design in-leakage assumption.

- Effective maintenance practices are in place to assure that the boundary is maintained.
- The plant configuration control program evaluates the effect of modifications on CRE integrity.
- Few simple modifications affecting CRH have been performed.
- Retesting of the CRE integrity demonstrates little change in in-leakage from the baseline test.

As an example, CRE integrity that meets the following attributes may require an assessment frequency of 18 months (and requires retesting):

- Frequent problems with CRE breach control program.
- The CRE has a large number of vulnerabilities to in-leakage.
- The measured CRE boundary differential pressure is less than the recommended margin allowed.
- The measured CRE in-leakage is close to the design in-leakage assumption.
- Maintenance practices do not ensure that the CRE boundary is maintained.
- The plant configuration control program lacks guidance to evaluate the effect of modifications on CRE integrity.
- Numerous or complex modifications to CR systems or the CRE barrier have been performed.

9.3.3.3 Determine Need to Test

If the CRE periodic assessment indicates that the actual or potential CRE in-leakage has significantly degraded, then retesting may be required. Additionally, testing may be required where the assessment alone cannot provide reasonable assurance that the CRE in-leakage has not degraded or where changes (such as modifications) to the CRE have invalidated previous testing. For example, plants with a large number of in-leakage vulnerabilities and/or a small in-leakage margin may periodically perform testing to ensure that the actual in-leakage has not increased.

9.3.4 Toxic Gas Evaluation

Each plant typically performed an evaluation of toxic gas sources in the vicinity of the plant during its initial licensing process and in response to TMI Action Item III.D.3.4. Many licensees have reviewed their situations more recently and may have even verified that there are no toxic gas sources that pose a challenge to the habitability of their control room. While the use of toxic gases on-site is subject to programmatic controls, fixed off-site and transportation sources are beyond the direct control of the licensee. It is recommended that each licensee periodically perform a review of these sources in accordance with Appendix G.

Each licensee should establish a frequency for these periodic assessments, based on the number, size and type of industrial and transportation activities in the vicinity of the plant and regional and local changes in land use. New commercial facilities or expansion/changes to existing facilities within 5 miles of the plant may present new potential threats to CRH. Likewise, increases in traffic or changes in nearby transportation routes (waterways, roads, rail lines) may also challenge design assumptions. The periodicity of reassessments due to changing toxic gas sources should be plant specific, based upon on-site and off-site challenges. It is suggested that licensees with plants in industrialized areas perform such an assessment every 5 years; others should consider an assessment frequency of at least every 10 years.

9.4 Configuration Control

There are many changes that can take place on-site under plant control that can have an impact on CR habitability. The intent of the guidance in this section is to ensure that plant controls recognize the potential impact of a change on control room habitability. Licencees are encouraged to review their existing programs and consider modifying or establishing new programs, if appropriate, to address the control aspects discussed below.

9.4.1 CRE Boundary / Breach Control

Each plant should have CRE boundary controls in place. Appendix K contains the guidance for establishing those controls if they do not already exist. The controls assure that boundary breaches are recognized, that uncontrolled breaches to the CRE do not exist and that known breaches do not result in an unanalyzed condition.

9.4.2 Procedure Control

10CFR50, Appendix B Criterion V requires that each licensee establish its procedure controls. It is recommended that each plant review its existing controls to assure that potential CR integrity issues are recognized and appropriately considered when revising procedures.

9.4.3 Toxic Chemical Control

Each licensee should ensure that its chemical controls include a review of new chemicals brought on-site and that this review consider the impact of a potential release on the control room. It is recommended the controls also provide guidance regarding acceptable quantities or container sizes for chemicals approved for use on-site.

9.4.4 Design Change Control

10CFR50, Appendix B Criterion III requires that each licensee establish design controls. These controls ensure that the design bases are appropriately incorporated into the design and operation of the plant. The controls also ensure that design changes, which include permanent and temporary modifications, are subject to similar controls. Each plant should understand the design of the CRE and habitability systems and ensure that they control changes to the design. In addition, appropriate post-modification testing should ensure that safety analyses assumptions remain valid. It is recommended that the CR HVAC system engineer be familiar with habitability issues and review each related modification package.

9.4.5 Safety Analyses Control

Safety analyses are typically covered as part of a plant's design controls. The safety analyses typically include various assumptions about the CRE for CRH purposes. Examples are:

- In-leakage
- Change in release location
- Quantity of release
- System isolation characteristics
- Assumed accident source term
- Operator action assumptions.

The various assumptions are addressed in more detail in Appendices C, D and G.

Changes in the inputs and assumptions to these analyses can affect the integrity of the CRE. The licensee's engineering department should establish good communications to ensure system features are appropriately modeled and the potential impact of system changes on the licensing analyses is recognized.

9.5 Training

The complexity and breadth of CRH warrants training of personnel. It is recommended that a training needs analysis be performed to assess whether operation, maintenance and engineering support personnel understand the bases for the CRE integrity program and issues that influence control room habitability. The guidance in this document along with plant specific information will form a good basis for the training material.

9.6 Testing

CRE in-leakage testing will be required to establish a baseline in-leakage measurement and may be required from time to time as part of the periodic assessment of CRE integrity. Guidance on CRE testing is provided in Appendix I and in the various industry standards referenced in Table I-1. Testing decisions made during the periodic reassessment and any test results obtained must be documented and retained as part of the CRH Program records.

10 REFERENCES

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APPENDIX A

LICENSING BASIS HISTORY

This appendix provides an overview of the control room habitability regulatory and licensing history.

1. ORIGIN OF THE CONTROL ROOM GENERAL DESIGN CRITERIA AND EARLY REGULATORY GUIDANCE.

In February 1971, the Atomic Energy Commission published Appendix A, *General Design Criteria (GDC) for Nuclear Power Plants* to 10CFR50. 10CFR50.34(a)(3)(i) requires an applicant for a construction permit to describe the preliminary design of the facility including the principal design criteria in a preliminary Safety Analysis Report (PSAR). This paragraph includes a reference to Appendix A as establishing the minimum requirements. Criterion 19 (GDC 19), *Control Room,* provides for a control room, alternative shutdown station(s) and habitability requirements. GDC 19, in part, requires:

> *"Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident."*

Between 1965 and 1971, the NRC staff worked on issuing the final version of the GDCs. The control room criterion was variously numbered as GDC 11, 13, 17 and finally, 19. There were several draft versions and much coordination between the Commission, the staff, and the Advisory Committee on Reactor Safeguards (ACRS). In June 1967, the Commission published a draft of the GDCs in the *Federal Register* for public comment and interim guidance. Applicants for construction permits and operating licenses during this period may have referenced it in their PSARs and FSARs. Many licensees were required to meet the draft GDC on control room habitability as a condition for receiving their construction permit and/or their operating license.

While the GDCs were under development, applicants proposed, and the staff approved, various criteria for the control room. As an example, at one plant the NRC approved the criterion of 10 percent of the 10 CFR Part 100, §100.11 dose guidelines.

In the early 1970's, K. Murphy and K. Campe presented a method for evaluating radiological events in the control room. Additional information can be found in a 1974 paper by Murphy and Campe[2]. In 1974 and 1975, NRC Regulatory Guides 1.78 and 1.95 were issued to provide direction on the protection of the control room operator from accidental releases of hazardous chemicals or chlorine gas respectively.

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2 K.G. Murphy and K.M. Campe, *Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19, In Proceeding of 13th AEC Air Cleaning Conference, San Francisco, CA, CONF-740807,* U.S. Atomic Energy Commission, 1974.

2. THREE MILE ISLAND ACCIDENT EFFECT ON CONTROL ROOM HABITABILITY REQUIREMENTS

As a result of the accident at Three Mile Island the NRC staff developed a number of proposed actions to be implemented on operating reactors and on plants under construction. These actions were presented in NUREG-0660, TMI-2 Action Plan. In October 1980, NUREG-0737, *Clarification of TMI Action Plan Requirements*, was published. NUREG-0737 contained all TMI-related items approved for implementation by the Commission as of October 31, 1980. The actions in NUREG-0737 were applicable to operating reactors and applicants for operating licenses. The letter that transmitted NUREG-0737 was addressed to all licensees of operating plants, and applicants for operating licenses and holders of construction permits. The letter in NUREG-0737 stated that the staff "...expected the requirements contained herein will be met." Pursuant to 10CFR50.54(f), operating reactor licensees were to confirm that the implementation dates in Enclosure 1 of NUREG-0737 would be met. If they could not, a revised date was to be provided along with a justification for the delay, a proposed revised date for completion and any planned safety actions during the interim.

The Standard Review Plan (NUREG-0800), Revision 1 was issued by the NRC in July 1981. The Standard Review Plan (SRP) provides standard regulatory acceptance guidance to the NRC staff for review and approval of Licensee Safety Analysis Reports. The SRP identified that the limiting design basis accident (DBA) for CRH is the loss of coolant accident. However, other DBAs were to be reviewed to determine whether they could be more limiting. Licensees were to provide assurance that the habitability systems will operate under all postulated conditions (DBA) to permit the control room operators to remain in the control room to take appropriate actions required by GDC 19. Where modifications were needed for compliance with CRH requirements, a schedule for completion of these modifications was required. Sometimes modifications and other CRH actions were deferred pending future resolution of certain regulatory issues such as the alternative source term (10CFR50.67).

In May 1982, Generic Letter 82-10 was issued, that requested licensees to implement on a timely basis those TMI Action Items from NUREG-0737 which had not been addressed by Generic Letter 82-05. The Enclosure to Generic Letter 82-10 identified those items for which a schedule needed to be established or, if a schedule had been previously submitted, a reconfirmation of those schedule dates. TMI Action Item III.D.3.4, Control Room Habitability Requirements was in that Enclosure. In March 1983, the NRC issued an order to each reactor facility confirming licensee's commitment to post-TMI related issues. The order required each licensee to implement and maintain the specific items described in the Attachments to the Order in the manner described in the licensee's submittal noted in the Order.

Two classes of licensees were identified in item III.D.3.4.

- Licensees with control rooms that meet the guidance of the SRP needed only to describe their basis for determining that the guidelines were met.

- Licensees with control rooms that did not meet the guidelines of the SRP were required to analyze the control room exposures and submit the results.

3. Reviews of Control Room Habitability in the 1980's

Two issues related to CRH were identified by the ACRS in the early 1980s. These issues, which are discussed in NUREG-0933, are:

• GSI B-66, *Control Room Infiltration Measurements*, which identified that a key parameter affecting control room habitability is the magnitude of control room air infiltration rates.

• GSI 83, *Control Room Habitability*, which identified that loss of control room habitability following an accidental release of external airborne toxic or radioactive material or smoke can impair or cause loss of the control room operators capability to safely control the reactor.

The ACRS issued a letter to the Commission, on August 18, 1982, which identified a wide range of deficiencies in the maintenance and testing of engineered safety features designed to maintain control room habitability.

These ACRS concerns encompassed both plant licensing review and operations and inspection activities.

In January 1983, the NRC staff responded to the ACRS concerns and recommended increased training of NRC and licensee personnel in inspection and testing of control room habitability systems. The staff also provided a profile of control room HVAC system component failures based on an analysis of Licensee Event Reports from 1977 through mid-1982. On April 28, 1983, Nuclear Reactor Regulation (NRR) and Office of Inspection & Enforcement (OIE) representatives met with the ACRS Subcommittee on Reactor Radiological Effects to discuss the staff response. Based on the accomplishments above, GSI B-66 was considered resolved.

In May 1983, the ACRS issued a letter to the Executive Director of Operations (EDO) that expressed continuing concerns about control room habitability and provided both general and specific comments and recommendations for further staff evaluation. This basically defined GSI 83. In July 1983, NRR transmitted to the EDO a joint NRR/OIE proposal for evaluating the ACRS comments and recommendations and the adequacy of the control room habitability licensing review process and criteria. In August 1983, the EDO indicated agreement with the proposal and directed NRR to coordinate with OIE and the NRC Regional Offices to complete the program and submit a report to the EDO by June 1, 1984. In September 1983, NRR established a Control Room Habitability Working Group and a Steering Group for conducting and guiding the proposed review. The Control Room Habitability Working Group was expected to identify any recommended actions that would correct significant deficiencies in control room habitability design, installation, test or maintenance.

Following issuance of NUREG/CR-4960, it was recognized that the methodology used to evaluate control room habitability system design needed improvement. Accordingly, the NRC staff initiated activities to develop:

- improved methods for calculating control room dose and exposure levels,
- improved meteorological models for use in control room habitability calculations and
- revised exposure limits to toxic gases for control room operators.

The results of the improved methods were documented in NUREG/CR-5669 and NUREG/CR-6210. The HABIT Code was developed to provide an integrated code package for evaluating control room habitability. In 2000, the NRC issued a new regulation (10 CFR 50.67) allowing licensees to voluntarily request license amendments to revise their design basis to use alternate source term information in radiological consequence assessments, including those for control room habitability.

As recommended by the ACRS, the staff was expected to consider National Institution for Occupational Safety and Health recommendations for toxic chemicals in its revision of Regulatory Guide 1.78.

4. Tracer Gas Testing and the Evolution of Industry Activities

As of the date of this document, approximately 25 percent of the control rooms have been tracer gas testing to determine the amount of in-leakage into the control room envelope. The NRC reported early testing results at a July 16, 1998, public meeting on control room habitability. The testing data indicated that actual in-leakage was much greater than the amount assumed in control room habitability analyses. Licensees embarked on sealing programs, design improvements and/or revision to dose consequence analyses to ensure regulatory requirements were met.

NUREG/CP-0167, *Proceedings of the 25th DOE/NRC Nuclear Air Cleaning and Treatment Conference,* reported on control room envelope reconstitution efforts at one nuclear power plant and control room air in-leakage testing results at two nuclear power plants. Some of the conclusions from these reports were:

- Tracer gas testing was instrumental in definition and quantification of unfiltered leakage paths and represented documented measured in-leakage rates. The constant injection tracer technique was considered the most useful method.
- Well-managed sealing efforts are instrumental for assuring control room integrity.
- Proper airflow balancing is essential to obtaining control room envelope and adjacent area HVAC system design basis.

Following the July 1998 public meeting with NEI, utility representatives and representatives from the Nuclear HVAC Users Group, the NRC staff agreed to work with the industry to resolve issues regarding control room habitability.

NEI agreed to take the lead. This document, NEI 99-03, presents the results of a joint industry and NRC effort to develop guidance to address CRH.

5. Revision to General Design Criterion 19

In conjunction with the January 2000 issuance of the Alternative Source Term regulation, 10CFR50.67, GDC-19 was revised to allow licensees to use a dose criterion of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) when implementing an alternative source term. Regulatory Guide 1.183, *Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors,* was issued in July 2000 to provide guidance on implementing an alternative source term.

APPENDIX B

CONTROL ROOM HABITABILITY REGULATORY INFORMATION

1. REGULATORY REQUIREMENTS

General Design Criterion (GDC) 19 of Appendix A to 10CFR50 is the controlling requirement for control room habitability (CRH). As discussed in Appendix A, plants licensed or issued construction permits before 1971 may not be committed to GDC 19. The text of this criterion, as amended in December 1999 with the issuance of 10CFR50.67, is provided below:

Criterion 19-Control room. *A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.*

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses under part 52 of this chapter who do not reference a standard design certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

It is important to note that although GDC-19 provides a specific numeric criterion for only radiation doses, the scope of the GDC applies to other conditions that would prevent the requisite actions from being performed.

2. REGULATORY GUIDES

The control room is expected to be habitable following design basis events. The design basis events that establish the parameters for the design of control room features may vary from plant to plant. The Regulatory Guides listed below address various events and define some of the assumptions to be considered in the analysis and evaluation of each event.

- Regulatory Guide 1.3 - *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors*
- Regulatory Guide 1.4 - *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors*
- Regulatory Guide 1.5 - *Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors*
- Regulatory Guide 1.24 - *Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure*
- Regulatory Guide 1.25 - *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors*
- Regulatory Guide 1.52 - *Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants*
- Regulatory Guide 1.77 - *Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors*
- Regulatory Guide 1.78 - *Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release*
- Regulatory Guide 1.95 - *Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release*
- Regulatory Guide 1.98 - *Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor*
- Regulatory Guide 1.145 - *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*
- Regulatory Guide 1.183 - *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*

3. NUREGs

The technical reports listed below provide general information and results of research related to CRH.

- NUREG-0737 - *Clarification of TMI Action Plan Requirements*

  As noted in Appendix A, Generic Letter 82-10 required licensees to submit a report describing their efforts to address the TMI Action Plan Requirements and provide schedule commitments. The NRC issued orders confirming these commitments. The applicability of any NUREG-0737 item to a particular facility is dependent on the specific commitments made by the licensee.

NUREG-0737, Action Item III.D.3.4, *Control Room Habitability Requirements,* is one of the activities identified by the NRC after the Three Mile Island (TMI) accident. Each licensee and applicant were required to make a submittal addressing several questions regarding the design of their control room and habitability systems. On the basis of a review of these responses, the NRC typically documented the closeout of this TMI issue in a safety evaluation report (SER).

As a part of the CRH assessment effort, each utility should consider the response it provided to this issue, determine whether it still reflects the current design of the CRH features and confirms that there is a SER closing out the issue for its plant.

For a few plants, the NRC issued SERs that allowed some control room habitability issues to remain open due to pending anticipated NRC actions. The NRC has permitted some plants to use temporary compensatory measures, such as the use of self-contained breathing apparatus or potassium iodide pills to mitigate radiological dose after an accident.

With the issuance of the accident source term rule, 10CFR50.67, the NRC encouraged licensees to comply with TMI Action Item III.D.3.4 without compensatory measures.

- NUREG-0800 - *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*

The Standard Review Plan (SRP) was developed to provide guidance primarily for the NRC staff performing reviews of license applications. It was intended to better assure the quality and consistency of the review effort. It also offered a means of communication for information about regulatory matters and the license process.

The SRP was originally issued in 1975 as NUREG-75/087. The SRP was revised in its entirety in 1981 and republished as NUREG-0800. The new revision outlined the requirements and acceptance criteria for each topic and also incorporated new regulatory positions, including several derived since the Three Mile Island accident (see NUREG-0737, discussed above).

The SRP follows much the same outline as that for the Final Safety Analysis Report (at least for those plants that followed the standard format of Regulatory Guide 1.70). The key sections that relate to control room habitability include:

- Section 6.4 – Control Room Habitability Systems
- Section 9.4.1 – Control Room Ventilation Systems
- Section 11.3 – Waste Gas System Failure and Liquid Tank Rupture Events
- Chapter 15 sections – Accident Analysis

The SRP typically identified the applicable regulatory requirements, outlined the regulatory considerations and often provided acceptable values for analysis assumptions. The following excerpt from NUREG-0800, Section 6.4 is provided as an example:

The LOCA source terms determined from the EAB review in accordance with Appendix A to SRP Section 15.6.5 are routinely used to evaluate radiation levels external to the control room. .... Other DBAs [Design Basis Accidents] are reviewed to determine whether they might constitute a more severe hazard than the LOCA. If appropriate, an additional analysis is performed for the suspect DBAs.

- NUREG-0933 *A Prioritization of Generic Safety Issues*

NUREG-0933 presents the priority rankings for generic safety issues related to nuclear power plants. The purpose of these rankings is to assist in the timely and efficient allocation of NRC resources for the resolution of those safety issues that have a significant potential for reducing risk. Two issues related to CRH are Items GSI-B66 and GSI-83 (Appendix A). These issues are considered to be resolved with no new requirements for licensees to implement.

- NUREG-1465 *Accident Source Terms for Light Water Nuclear Power Plants*

In 1962, the U. S. Atomic Energy Commission published TID-14844 to specify the release of fission products from a postulated accident involving a substantial meltdown of the core. This source term was used by nearly all licensees to demonstrate compliance with the reactor siting criteria of 10CFR100 and has subsequently been used to estimate control room doses.

In 1995, the NRC published NUREG-1465 and provided more realistic estimates of the source term released from the core. This updated source term guidance was specifically applicable to future reactors. The Alternative Source Term Rule (10CFR50.67) was issued in December 1999 and provided for the implementation of the new source term insights of NUREG-1465 by currently licensed facilities. Regulatory Guide 1.183 provides a PWR and BWR alternative source term acceptable to the NRC staff and provides guidance regarding the attributes of an acceptable source term.

The NRC staff has also rebaselined a PWR and BWR using the NUREG-1465 source terms (SECY-98-154) and concluded the alternative source term need not be imposed on licensees because use of TID-14844 provides adequate protection of the public. The NRC concluded that voluntary application of the alternative source term by licensees of currently operating plants would be acceptable as an opportunity for burden reduction. Implementation must be approved by the NRC in an amendment to the plant operating license.

While not directly associated with the CRH issue, the alternative source term does offer an improved basis for a larger control room in-leakage value than initially assumed. The new source term, in conjunction with its switch to total effective dose equivalent acceptance criteria, may yield acceptable calculated dose consequences for the postulated accidents in a plants' licensing basis. Appendix C provides additional details concerning the use of the alternative source term.

- NUREG/CP-0167 *25th DOE/NRC Nuclear Air Cleaning and Treatment Conference*

  NUREG/CP-0167 contains papers presented at the conference without associated comments. Major topics included control room safeguards. For example, one session was "HVAC Systems for Control Rooms and Other Nuclear Facilities."

- NUREG/CR-4960 *Control Room Habitability Survey of Licensed Commercial Nuclear Power Generating Station*

  NUREG/CR-4960 presents the results of a survey of 12 plants regarding the design of their systems used for control room habitability. The survey was conducted from 1986 to 1988 and was published in September 1988. The observations may offer insights to licensees preparing to assess the integrity and effectiveness of their own control room envelope.

- NUREG/CR-6210 *Computer Codes for Evaluation of Control Room Habitability (HABIT)*

  NUREG/CR-6210 describes the HABIT package of computer codes designed to be used for the evaluation of control room habitability in the event of an accidental release of toxic chemicals or radioactive materials.

  HABIT is an integrated package of several programs that previously needed to be run separately and required considerable use intervention. Two of these modules, EXTRAN and CHEM, are used for estimating chemical exposures. EXTRAN determines the release rate of a chemical in the event of leaks or ruptures of liquid or gas tanks. It also uses a model that computes atmospheric dilution, including the effects of building wakes, to determine the chemical concentration arriving at the intake to the control room. CHEM models the dilution of the chemical by flows in the control room and determines the chemical exposure to control room personnel.

  Draft Regulatory Guide DG-1087 (Proposed Revision 1 to Regulatory Guide 1.78), *Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release*, endorses the use of EXTRAN to model the atmospheric transport of a released hazardous chemical as part of a licensee's toxic gas assessment. The use of EXTRAN as part of a toxic gas assessment is also discussed in Appendix G.

- NUREG/CR-6331, Rev. 1 *Atmospheric Relative Concentrations in Building Wakes (ARCON96)*

  NUREG/CR-6331 describes the Atmospheric Relative Concentration in Building Wakes (ARCON96) computer code. ARCON96 is an atmospheric dispersion code intended for use in control room habitability assessments. The code uses hourly meteorological data and refined methods for estimating dispersion in the vicinity of buildings to calculate relative concentrations at control room air intakes that would be exceeded no more than 5 percent of the time. These concentrations are calculated for averaging periods ranging from one hour to 30 days in duration.

The use of ARCON96 as part of a radiological dose analysis is discussed in Appendix D.

- NUREG/CR-6604 *RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation*

  NUREG/CR-6604, documents the RADTRAD computer code developed for the NRC to estimate transport and removal of radionuclides and dose at selected receptors. The code can be used to estimate releases using various source terms. Additionally, the code can account for a reduction in the quantity of radioactive material due to containment sprays, natural deposition, filters and other natural and engineered safety features.

4. Inspection and Enforcement Notices (IEN) and Information Notices (IN)

The following notices provide information regarding designs or events that had an identified impact on control room habitability.

- IEN 83-41 – *Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment*
- IEN 83-62 – *Failure of Redundant Toxic Gas Detectors Positioned at Control Room Ventilation Air Intakes*
- IEN 83-69 – *Improperly Installed Fire Dampers at Nuclear Power Plants*
- IEN 86-76 – *Problems Noted in Control Room Emergency Ventilation Systems*
- IN 88-61 – *Control Room Habitability - Recent Reviews of Operating Experience*
- IN 89-44 – *Hydrogen Storage on the Roof of the Control Room*
- IN 91-56 – *Potential Radioactive Leakage to Tank Vented to Atmosphere*
- IN 92-18 – *Potential for Loss of Remote Shutdown Capability during a Control Room Fire*
- IN 92-32 – *Problems Identified with Emergency Ventilation Systems for Near-Site (within 10 Miles) Emergency Operations Facilities & Technical Support Centers*
- IN 93-06 – *Potential Bypass Leakage Paths Around Filters Installed in Ventilation Systems*
- IN 97-01 – *Improper Electrical Grounding Results in Simultaneous Fires in the Control Room and the Safe Shutdown Equipment Room*

- IN 97-79 – *Potential Inconsistency in the Assessment of the Radiological Consequences of a Main Steam Line Break Associated With the Implementation of Steam Generator Tube Alternate Repair Criteria*
- IN 97-82 – *Inadvertent Control Room Halon Actuation Due to a Camera Flash*
- IN 99-05 – *Inadvertent Discharge of Carbon Dioxide Fire Protection System and Gas Migration*

5. Regulatory Issue Summaries
   - RIS 2001-09 – *Control of Hazard Business*

6. Generic Letters
   - GL 82-05 – *Post TMI Requirements*
   - GL 82-10 – *Post-TMI Lessons Learned*
   - GL-99-02 – *Laboratory Testing of Nuclear-Grade Activated Charcoal*

APPENDIX C

CRH DOSE ANALYSIS: REGULATORY ENHANCEMENTS

1. PURPOSE AND SCOPE

This appendix summarizes dose analysis information provided by regulatory guides and NUREG-0800, the Standard Review Plan. Furthermore, it incorporates improvements in analytical methods and assumptions based on CRH Task Force discussions with the NRC staff.[3A]

This appendix describes radiological analysis assumptions for use in conjunction with the Source Term Technical Information Document (TID) 14844, *Calculation of Distance Factors for Power and Test Reactor Sites*. In 10 CFR Part 50, *Domestic Licensing of Production and Utilization Facilities*, Section 50.34, *Contents of Applications; Technical Information*, requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. Applicants are also required by 10 CFR 50.34 to provide an analysis of the proposed site. 10 CFR Part 100, *Reactor Site Criteria*, Section 100.11, *Determination of Exclusion Area, Low Population Zone, and Population Center Distance*, provides criteria for evaluating the radiological aspects of the proposed site. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based upon a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

This appendix provides guidance for performing control room dose calculations in support of control room habitability analyses for design basis radiological accidents. As described in Sections 3.4 and 6.2 of the main body of this document, this includes an assessment of each accident within the licensing basis of the facility. The purpose of this assessment is to determine the limiting event with respect to control room dose. Some licensees have previously evaluated the control room dose only for the design basis accident (DBA) Loss of Coolant Accident (LOCA), which is typically the limiting event for off-site radiological releases. The DBA LOCA is generally the large break LOCA event analysis with the radioactive source term specified in Regulatory Guide 1.3 or 1.4. Other events may be analyzed as part of the design basis accident evaluation for the facility. Although these events may have been shown to be non-limiting with respect to offsite dose, control room dose analyses should be performed for these events to identify the limiting event for the GDC 19 control room dose design criterion.

The DBAs that should be examined are those described in the final safety analysis report (FSAR) or licensee commitments. The assumptions used in the analyses of these events

3 Endnotes are designated by uppercase letters. Section 8 of this appendix provides the endnotes with a statement of their purpose.

remain as stated in the plant licensing basis. For example, assumptions in the licensing basis analyses regarding concurrent loss of off-site power for accidents do not change.

TID-14844 is cited in 10 CFR Part 100 as a source of further guidance on these analyses. Although initially used only for siting evaluations, the TID-14844 source term has been used in other design basis applications, such as environmental qualification of equipment under 10 CFR 50.49, *Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants*, and in some requirements related to Three Mile Island (TMI) as stated in NUREG-0737, *Clarification of TMI Action Plan Requirements*. The analyses and evaluations required by 10 CFR 50.34 for an operating license is documented in the facility FSAR. Fundamental assumptions that are design inputs, including the source term, are to be included in the FSAR and become part of the facility design basis. The DBAs were not intended to be actual event sequences, but rather, were intended to be surrogates to enable deterministic evaluation of the response of a facility's engineered safety features. These accident analyses are intentionally conservative to compensate for known uncertainties in accident progression, fission product transport and atmospheric dispersion. [B]

2. USE OF ALTERNATIVE SOURCE TERMS

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude and chemical form of fission product releases from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, *Accident Source Terms for Light-Water Nuclear Power Plants*. NUREG-1465 used this research to provide estimates of the accident source term that were more physically based and that could be applied to the design of future light-water power reactors. NUREG-1465 presents a representative accident source term for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment. The NRC staff considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety. Operating reactors licensed under that approach would not be required to reanalyze accidents using the revised source terms. The NRC staff also determined that some licensees might wish to use an alternative source term (AST) in analyses to support cost-beneficial licensing actions. Therefore, the NRC staff initiated several actions to provide a regulatory basis for operating reactors to use an AST in design basis analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67 and Regulatory Guide 1.183.

Licensees planning to use an AST methodology should refer to Regulatory Guide 1.183.

3. ACCIDENT SOURCE TERM FOR TID-14844 ANALYSES

This section describes an accident source term for TID-14844 analyses. It provides guidance on the fission product inventory, release fractions, timing of the release, radionuclide composition, chemical form and the fuel damage for DBAs. In general, the content of the guidance in this section follows the form of Regulatory Guide 1.183. [A]

3.1 Fission Product Inventory

The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum achievable, full-power operational history of the core with the assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty.[4] Fuel enrichment values and tolerances and the licensed fuel burnup range should be examined to maximize fission product inventory. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP. Core inventory factors (Ci/MWt) provided in TID-14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.

For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. For events where it is appropriate to consider differences in power level across the core, analyses should incorporate peaking factors from the facility's core operating limits report or technical specifications in determining the inventory of the damaged rods.

No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full, rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shut down, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.

3.2 Release Fractions[5]

The core inventory release fractions, by radionuclide group, for the DBA LOCAs are listed in Table 1 for BWRs and PWRs. These fractions are applied to the equilibrium core inventory described in Section 3.1.

For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 2. The release fractions from Table 2 are used in conjunction with the calculated fission product inventory as described in Section

4 The uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02. A value lower than 1.02, but not less than the licensed power level may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation level. See Federal Register Notice Volume 65, Number 106, June 1, 2000, for details.

5 The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide fuel.

3.1. For non-LOCA DBAs where fuel melt is postulated, the core inventory release fractions, by radionuclide group, are listed in Table 1 for BWRs and PWRs.

Table 1

BWR AND PWR CORE INVENTORY FRACTION RELEASED INTO CONTAINMENT ATMOSPHERE

Group	Release Fraction
Noble Gases	1.0
Iodines[6]	0.5
Other Nuclides	0.0

Table 2[7]

NON-LOCA FRACTION OF FISSION PRODUCT INVENTORY IN GAP

Group	Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Iodines	0.05
Other Nuclides	0.0

3.3 Timing of Release Phases

For LOCA analyses, the core activity released is assumed to be immediately available for release from the containment. For non-LOCA DBAs in which fuel damage is projected,

6 If containment sprays are not modeled mechanistically, such as in SRP Section 6.5.2, revision 2, one half of the equilibrium radioactive iodine inventory may be assumed to be deposited on the walls of the containment. The net value of core inventory available for release from the containment would, therefore, be 0.25 for a non-mechanistic spray representation. Please note that SRP Section 6.5.2, revision 2 erroneously states that 25 percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for the leakage from the primary reactor system. This value should be 50 percent of the equilibrium radioactive iodine inventory. Revision 2 erroneously accounted twice for the iodine deposited on the wall of the containment.

7 The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for burnup exceeding 54,000 MWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10 percent for iodines and noble gases.

the activity available for release from the fuel is assumed to be immediately available for release from the containment or the building where the fuel is damaged.

3.4 Radionuclide Composition

Table 3 lists the elements in each radionuclide group that should be considered in design basis analyses.

Table 3

RADIONUCLIDE GROUPS

Group	Elements
Noble Gases	Xe, Kr
Iodines	I

3.5 Chemical Form

Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated LOCA, 5 percent should be assumed to be particulate iodine, 91 percent elemental iodine and 4 percent organic iodide. This includes releases from the gap and the fuel pellets. The same chemical form is assumed in releases from the fuel pins through the RCS in non-LOCA DBAs other than fuel handling accidents (FHAs). The chemical form assumed in releases from fuel pins in FHAs is addressed in Section 6.2 of this appendix. The transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific descriptions in Section 6 of this appendix provide additional details.

3.6 Fuel Damage in Non-LOCA DBAs

The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.

The amount of fuel damage caused by a FHA is addressed in Section 6.2 of this appendix.

4. Dose Consequences Calculation Methodology

The guidance of this section applies to all dose calculations performed with TID-14844 source terms. Section 4.1 describes the determination of consequences for off-site receptor evaluations. Section 4.2 describes comparable assumptions for control room personnel evaluations. Section 4.3 presents the acceptance criteria for each scenario described in Section 6 of this appendix. Although the main focus of this appendix is the analysis of CR dose, the guidance in this appendix, in general, may also be applied to the evaluation of off-site dose.

4.1 Off-site Dose Consequences

The following assumptions should be used in determining the whole body and thyroid dose for persons located at or beyond the exclusion area boundary (EAB):

4.1.1 The dose conversion factors for inhalation of radioactive material may be derived from the data provided in ICRP Publication 30, *Limits for Intakes of Radionuclides by Workers*. Table 2.1 of Federal Guidance Report 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, provides tables of conversion factors acceptable to the NRC staff. C

4.1.2 For the first 8 hours, the breathing rate of persons off-site should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the breathing rate should be assumed to be 2.3×10^{-4} cubic meters per second.

4.1.3 The whole body dose should be calculated assuming submergence in a semi-infinite cloud with appropriate credit for attenuation by body tissue. Table III.1 of Federal Guidance Report 12, *External Exposure to Radionuclides in Air, Water, and Soil*, provides external dose conversion factors acceptable to the NRC staff.

4.1.4 Doses determined for the most limiting receptor at the EAB and at the outer boundary of the low population zone (LPZ) should be used in determining compliance with the dose criteria in 10 CFR 100.11.

4.1.5 No correction should be made for depletion of the effluent plume by deposition on the ground.

4.2 Control Room Dose Consequences

The following guidance should be used for calculating doses to persons located in the control room:

4.2.1 The dose analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include:

- Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,
- Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope,
- Radiation shine from the external radioactive plume released from the facility,
- Radiation shine from radioactive material in the reactor containment,
- Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.

4.2.2 The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport and release assumptions used for determining the EAB and the LPZ dose values.

4.2.3 The models used to transport radioactive material into and through the control room[8], and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.

4.2.4 Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to SRP Section 6.5.1, *ESF Atmospheric Cleanup System,* to Regulatory Guide 1.52, *Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants*, and to Generic Letter 99-02 (Refs. 14 and 15), for guidance. The control room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by

8 The iodine protection factor (IPF) methodology of Reference 11 may not be adequately conservative for all DBAs and control room arrangements since it models a steady state control room condition. Since many analysis parameters change over the duration of the event, the IPF methodology should only be used with caution. The NRC computer codes HABIT (Ref. 12) and RADTRAD (Ref. 13) incorporate suitable methodologies.

engineered safety feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.

4.2.5 Credit should generally not be taken for the use of personal protective equipment or the use of potassium iodide as a thyroid prophylaxis. Additional guidance for use of such equipment for temporary compensatory measures is provided in Appendix F. Deviations may be considered on a case-by-case basis.

4.2.6 The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100 percent of the time during the first 24 hours after the event, 60 percent of the time between 1 and 4 days, and 40 percent of the time from 4 days to 30 days[9]. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.

4.2.7 Control room doses should be calculated using dose conversion factors identified in Section 4.1 above for use in off-site dose analyses. The whole body dose from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, WB_∞, to a finite cloud dose, WB_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room.

$$WB_{finite} = \frac{WB_\infty V^{0.338}}{1173}$$

This correction is not applied to the beta skin dose estimates, as the range of beta particles in air is less than the typical control room dimensions. The skin dose conversion factors (DCFs) presented in Federal Guidance Report 12 are based on both photon and beta emissions. Without the geometry correction, the photon dose component will be overestimated. If the geometry correction is included, the beta component will be underestimated. DOE/EH-0070 tabulates the beta and photon skin DCFs separately.

4.3 Acceptance Criteria

The radiological criteria for the EAB and the outer boundary of the LPZ and the criteria for the control room are specified in 10 CFR 100.11 and 10 CFR 50, respectively. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of

9 This occupancy is modeled in the χ/Q values determined in Reference 11 and should not be credited twice. The ARCON96 code (Ref. 16) does not incorporate these occupancy assumptions, making it necessary to apply this correction in the dose calculations.

occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 4.

The acceptance criteria for the various NUREG-0737 items generally reference General Design Criteria 19 (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. Standard Review Plan 6.4 allows for higher beta dose exposure provided that protective clothing is used. For facilities applying for, or having received, approval for the use of an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).

Table 4 [D]

Accident Dose Acceptance Criteria (rem)						
	Control Room			EAB / LPZ		
Accident or Case	**Thyroid**	**Whole Body**	**Skin**	**Thyroid**	**Whole Body**	**Analysis Release Duration**
LOCA (BWR & PWR)	50	5	30	300	25	30 days for containment, ECCS and MSIV (BWR) leakage
BWR MAIN STEAM LINE BREAK ACCIDENT						Instantaneous puff
• Fuel Damage or Pre-incident Spike	50	5	30	300	25	
• Equilibrium Iodine Activity	50	5	30	30	2.5	
BWR CONTROL ROD DROP ACCIDENT	50	5	30	75	6.3	24 hours
PWR STEAM GENERATOR TUBE RUPTURE[10]						Affected SG: time to isolate; Unaffected SG(s): until cold shutdown is established
• Fuel Damage or Pre-incident Spike	50	5	30	300	25	
• Coincident Iodine Spike	50	5	30	30	2.5	
PWR MAIN STEAM LINE BREAK ACCIDENT						Until cold shutdown is established
• Fuel Damage or Pre-incident Spike	50	5	30	300	25	
• Coincident Iodine Spike	50	5	30	30	2.5	
PWR LOCKED ROTOR ACCIDENT	50	5	30	30	2.5	Until cold shutdown is established
PWR ROD EJECTION ACCIDENT	50	5	30	75	6.3	30 days for containment pathway; until cold shutdown is established for secondary pathway
FUEL HANDLING ACCIDENT	50	5	30	75	6.3	Off-site Pathway: 2 hours Control Room: 30 days

The column labeled Analysis Release Duration is a summary of the assumed radioactivity release durations identified in Section 6 of this appendix. Refer to Section 6 for complete descriptions of the release pathways and durations.

10 For PWRs with steam generator alternate repair criteria, different dose criteria may apply to steam generator tube rupture and main steam line break analyses as described in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," USNRC, December 1998 (Ref. 18).

5. ANALYSIS ASSUMPTIONS AND METHODOLOGY

5.1 General Considerations

5.1.1 Analysis Quality

These analyses should be prepared, reviewed and maintained in accordance with quality assurance programs that comply with Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, to 10 CFR Part 50.

5.1.2 Credit for Engineered Safety Features

Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures.

An adequate failure mode analysis is to be performed to justify the selection of the most limiting single active failure for use in the radiological consequence analysis of each accident to be evaluated. When required in a plant's licensing basis, coincident loss of off-site power is assumed at the time of the accident.

5.1.3 Assignment of Numeric Input Values

The numeric values that are chosen as inputs to the analyses should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be non-conservative in another portion of the same analysis. For example, assuming minimum containment, system spray flow is usually conservative for estimating iodine scrubbing, but in many cases may be non-conservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by technical specifications, the value used in the analysis should be that specified in the technical specifications.[11] If a range of values or a tolerance band is specified, the value that would result in a conservative postulated dose should be used. If the parameter is based on the results of less frequent surveillance testing, e.g., steam generator nondestructive testing, consideration should be given to the degradation that may occur between periodic tests in establishing the analysis value.

11 Note that for some parameters, the technical specification value may be adjusted for analysis purposes by factors provided in other regulatory guidance. For example, ESF filter efficiencies are based on the guidance in Regulatory Guide 1.52 (Ref. 14) and in Generic Letter 99-02 (Ref. 15) rather than the surveillance test criteria in the technical specifications. Generally, these adjustments address potential changes in the parameter between scheduled surveillance tests.

5.1.4 Applicability of Current Licensing Basis

Applicability of current licensing basis is discussed in Sections 4.5 and 6.2 of the main body of this document.

5.2 Meteorology Assumptions

Atmospheric dispersion values (χ/Q) for the EAB, the LPZ and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses described in this appendix. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, and the Murphy-Campe paper, *Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19* (Refs. 11, 19, 20, and 21).

References 11 and 21 should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. For TID source term applications, where immediate release is assumed, this will be the zero to 2-hour exposure period. The NRC computer code PAVAN implements Regulatory Guide 1.145 and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96[12] is generally acceptable to the NRC staff for use in determining control room χ/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility UFSAR should be used in generating accident χ/Q values. Additional guidance is provided in Regulatory Guide 1.23, *On-site Meteorological Programs*. All changes in χ/Q analysis methodology should be reviewed by the NRC staff.

Further guidance for performing atmospheric dispersion calculations in support of CRH analysis is provided in Appendix D.

12 The ARCON96 computer code contains processing options that may yield χ/Q values that are not sufficiently conservative for use in accident consequence assessments or may be incompatible with release points and ventilation intake configurations at particular sites. The applicability of these options and associated input parameters should be evaluated on a case-by-case basis. The assumptions made in the examples in the ARCON96 documentation are illustrative only and do not imply NRC staff acceptance of the methods or data used in the example.

6. CONTROL ROOM HABITABILITY ANALYSIS ASSUMPTIONS

6.1 Assumptions for Evaluating the Radiological Consequences of a LWR Loss of Coolant Accident

The assumptions discussed in this section are for evaluating the radiological consequences of LOCAs at light water reactors (LWRs). These assumptions supplement the guidance provided in Sections 1 through 5 of this appendix.

Appendix A, *General Design Criteria for Nuclear Power Plants*, to 10 CFR Part 50 defines LOCAs as those postulated accidents that result from a loss of coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the reactor coolant system are included. The LOCA, as with all DBAs, is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. Analyses are performed using a spectrum of break sizes to evaluate fuel and emergency core cooling system (ECCS) performance. With regard to radiological consequences, a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility.

6.1.1 Source Term Assumptions

Recommended assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Section 3 of this appendix.

6.1.2 Assumptions on Transport in Primary Containment

Recommended assumptions related to the transport, reduction, and release of radioactive material in and from the primary containment in PWRs or the drywell in BWRs are as follows:

6.1.2.1 The radioactivity released from the fuel should be assumed to initially mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs. This distribution should then be adjusted as a function of time if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell and the wetwell.

6.1.2.2 Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. An acceptable model for removal of iodine and particulates is described in Section 6.5.2, *Containment Spray as a Fission Product Cleanup System*, of the Standard Review Plan (SRP), NUREG-0800.

6.1.2.3 Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with SRP Section 6.5.2[13] may be credited. An acceptable model for the removal of iodines is described in SRP Section 6.5.2.

The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed region volume per hour, unless other rates are justified. On a case by case basis containment mixing rates determined by the cooldown rate in the sprayed region and the buoyancy-driven flow that results may be considered.[E] The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90 percent of the volume and if adequate mixing of unsprayed compartments can be shown.

The SRP sets forth a maximum decontamination factor (DF) for elemental iodine, defined as the ratio of activity in the primary containment atmosphere when the sprays actuate to the activity remaining at some time after spray actuation. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached.

6.1.2.4 Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02.

6.1.2.5 Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs is given in SRP Section 6.5.5. For suppression pool solutions having pH less than 7, molecular iodine vapor should be conservatively assumed to evolve into the containment atmosphere. [F]

6.1.2.6 Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineered safety features not addressed above, should be evaluated on an individual case basis. See SRP Section 6.5.4.

6.1.2.7 The primary containment (drywell for BWRs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50 percent of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50 percent of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications.

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13 Note that SRP Section 6.5.2, revision 2 erroneously states that 25 percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for the leakage from the primary reactor system. This value should be 50 percent of the equilibrium radioactive iodine inventory. Revision 2 erroneously accounted twice for the iodine deposited on the wall of the containment.

6.1.2.8 If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100 percent of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered.

6.1.3 Assumptions on Dual Containments

For facilities with dual containment systems, the recommended assumptions related to the transport, reduction and release of radioactive material in and from the secondary containment or enclosure buildings are as follows.

6.1.3.1 Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.

6.1.3.2 Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.

6.1.3.3 The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5 percent of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded either 5 percent or 95 percent of the total number of hours in the data set, whichever is conservative for the intended use. If high temperatures are limiting, use that value exceeded by only 5 percent. If low temperatures are limiting, use that value exceeded by at least 95 percent.

6.1.3.4 Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Dilution credit from mixing, if evaluated to be applicable, will generally be limited to 50 percent This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.

6.1.3.5 Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.

6.1.3.6 Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02.

6.1.4 Assumptions on ESF System Leakage

ESF systems that recirculate sump water outside the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections and other similar components. This release source may also include leakage through valves isolating interfacing systems. The radiological consequences from the postulated leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are recommended for evaluating the consequences of leakage from ESF components outside the primary containment for BWRs and PWRs.

6.1.4.1 With the exception of noble gases, all the fission products released from the fuel to the containment should be assumed to mix instantaneously and homogeneously in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the start of the accident. As shown in Table 1 in Section 3 of this appendix, this applies to the release fraction of 50 percent of the core iodine inventory. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are non-conservative with regard to the buildup of sump activity.

6.1.4.2 The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737, would require declaring such systems inoperable. [G] The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., ECCS pump miniflow return to the refueling water storage tank.

6.1.4.3 With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.

6.1.4.4 If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment:

$$FF = \frac{h_{f_1} - h_{f_2}}{h_{fg}}$$

where h_{f_1} is the enthalpy of liquid at system design temperature and pressure, h_{f_2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F) and h_{fg} is the heat of vaporization at 212°F.

6.1.4.5 If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10 percent, the amount of iodine that becomes airborne should be assumed to be 10 percent of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.

6.1.4.6 Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 and Generic Letter 99-02.

6.1.5 Assumptions on Main Steam Isolation Valve Leakage in BWRs

For BWRs, the main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The radiological consequences from postulated MSIV leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are recommended for evaluating the consequences of MSIV leakage.

6.1.5.1 For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Section 3 of this Appendix). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.

6.1.5.2 All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50 percent of the maximum leak rate.

6.1.5.3 Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.

6.1.5.4 In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.1.5.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground-level release. Generally, holdup and dilution in the turbine building should not be assumed.

6.1.5.5 A reduction in MSIV releases, which is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake. The amount of reduction allowed will be evaluated on an individual case basis. References 25 and 26 provide guidance on acceptable models.

6.1.6 Assumption on Containment Purging

The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 and Generic Letter 99-02.

6.2 Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident

This section provides assumptions for evaluating the radiological consequences of a fuel handling accident at light water reactors. These assumptions supplement the guidance provided in Sections 3 through 5 of this appendix.

6.2.1 Source Term Assumptions

Recommended assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Section 3 of this appendix. The following assumptions also apply.

6.2.1.1 The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.

6.2.1.2 The fission product release from the breached fuel is based on Section 3.2 and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons and iodines.

6.2.1.3 The iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%).

6.2.1.4 A conservative approach to determining the quantity of radioactive material available for release from a fuel assembly is to assume that the assembly with the peak inventory is the one damaged. The fission product inventory for the peak assembly represents an upper limit value and is not expected to be exceeded. The inventory should be calculated assuming the maximum achievable operational power history at the end of core life immediately preceding shutdown. This inventory calculation should include appropriate assembly peaking factors. [H]

6.2.2 Pool Decontamination Factor

If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective DF of 200 [I] (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in DFs for elemental (99.75%) and organic iodine (0.25%) species results in the iodine above the water being composed of 44 percent elemental and 56 percent organic species. If the depth of water is not at least 23 feet, the DF will have to be determined on a case-by-case method. Proposed increases in the pool DF above 200 will need to address re-evolution of the scrubbed iodine species over the accident duration and should be supported by empirical data.

For release pressures greater than 1200 psig [J] or water depth less than 23 feet, the iodine DFs will be less than those assumed in this section and must be calculated on an individual basis using assumptions comparable in conservatism to those herein.

6.2.3 Noble Gases and Particulates

The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).

6.2.4 Fuel Handling Accidents Within the Fuel Building

For fuel handling accidents postulated to occur within the fuel building, the following assumptions apply.

6.2.4.1 The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period. The release rate is a function of the plant configuration but is generally assumed to be a linear or exponential function over this time period.

6.2.4.2 A reduction in the amount of radioactive material released from the fuel pool by ESF filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. 14 and 15). Delays in radiation detection, actuation of the ESF filtration system or diversion of ventilation flow to the ESF filtration system[14] should be determined and accounted for in the radioactivity release analyses.

6.2.4.3 The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.

6.2.5 Fuel Handling Accidents Within Containment

For fuel handling accidents postulated to occur within the containment, the following assumptions apply.

14 These analyses should consider the time for the radioactivity concentration to reach levels corresponding to the monitor setpoint, instrument line sampling time, detector response time, diversion damper alignment time and filter system actuation, as applicable.

6.2.5.1 If the containment is isolated[15] during fuel handling operations, no radiological consequences need to be analyzed.

6.2.5.2 If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.

6.2.5.3 If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[16] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period. The release rate is a function of the plant configuration but is generally assumed to be a linear or exponential function over this time period.

6.2.5.4 A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. 14 and 15). Delays in radiation detection, actuation of the ESF filtration system or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.

6.2.5.5 Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50 percent of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.

6.3 Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident

This section provides assumptions for evaluating the radiological consequences of a rod drop accident at BWR light-water reactors. These assumptions supplement the guidance provided in Sections 3 through 5 of this appendix.

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15 Containment *isolation* does not imply containment integrity as defined by technical specifications for non-shutdown modes. The term isolation is used here collectively to encompass both containment integrity and containment closure, typically in place during shutdown periods. To be credited in the analysis, the appropriate form of isolation should be addressed in technical specifications.

16 The staff will generally require that technical specifications allowing such operations include administrative controls to close the airlock, hatch or open penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.

6.3.1 Source Term Assumptions

Recommended assumptions regarding core inventory are provided in Section 3 of this appendix. For the BWR rod drop accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10 percent of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100 percent of the noble gases and 50 percent of the iodines contained in that fraction are released to the reactor coolant.

6.3.2 Coolant Activity Assumptions

If no or minimal[17] fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity (typically a pre-existing spike of 4 μCi/gm DE I-131) allowed by the technical specifications.

6.3.3 Transport, Reduction and Release Assumptions

The assumptions acceptable to the NRC staff that are related to the transport, reduction and release of radioactive material from the fuel and the reactor coolant are as follows.

6.3.3.1 The activity released from the fuel from the gap and/or fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.

6.3.3.2 Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.

6.3.3.3 Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10 percent of the iodine and 1 percent of the remaining radionuclides are assumed to reach the turbine and condensers.

6.3.3.4 Of the activity that reaches the turbine and condensers, 100 percent of the noble gases and 10 percent of the iodine are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground-level release at a rate of 1 percent per day[18] for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condensers may be assumed.

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17 Minimal fuel damage is defined as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining the dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

18 If there are forced flow paths from the turbine or condenser, such as unisolated mechanical vacuum pumps or unprocessed air ejectors, the leakage rate should be assumed to be the flow rate associated with the most limiting of these paths. Credit for collection and processing of releases, such as by off gas or standby gas treatment, will be considered on a case-by-case basis.

6.3.3.5 In lieu of the transport assumptions provided in paragraphs 6.3.3.2 through 6.3.3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first MSIV and considers MSIV closure time.

6.3.3.6 The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 5 percent particulate, 91 percent elemental and 4 percent organic.

6.4 Assumptions for Evaluating the Radiological Consequences of a BWR Main Steam Line Break Accident

This section provides assumptions for evaluating the radiological consequences of a main steam line accident at BWR light water reactors. These assumptions supplement the guidance provided in Sections 3 through 5 of this appendix.

6.4.1 Source Term Assumptions

Recommended assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Section 3 of this appendix. The release from the breached fuel is based on Section 3.2 and the estimate of the number of fuel rods breached.

6.4.2 Coolant Activity Assumptions

If no or minimal[19] fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specification. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the nuclear steam supply system vendor's standard technical specifications.

6.4.2.1 The concentration that is the maximum value (typically 4.0 μCi/gm DE I-131) permitted and corresponds to the conditions of an assumed pre-accident spike, and

6.4.2.2 The concentration that is the maximum equilibrium value (typically 0.2 μCi/gm DE I-131) permitted for continued full power operation.

6.4.3 Reactor Coolant Mixing Assumptions

The activity released from the fuel should be assumed to mix instantaneously and homogeneously in the reactor coolant. Noble gases should be assumed to enter the steam phase instantaneously.

19 Minimal fuel damage is defined as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining the dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

6.4.4 Transport, Reduction and Release Assumptions

Assumptions acceptable to the NRC staff related to the transport, reduction and release of radioactive material to the environment are as follows.

6.4.4.1 The main steam line isolation valves should be assumed to close in the maximum time allowed by technical specifications.

6.4.4.2 The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure.

6.4.4.3 All the radioactivity in the released coolant should be assumed to be released to the atmosphere instantaneously as a ground-level release. No credit should be assumed for plateout, holdup or dilution within facility buildings.

6.4.4.4 The iodine species released from the main steam line should be assumed to be 5 percent particulate, 91 percent elemental and 4 percent organic.

6.5 Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident

This section provides assumptions for evaluating the radiological consequences of a steam generator tube rupture accident at PWR light-water reactors.[20] These assumptions supplement the guidance provided in Sections 3 through 5 of this appendix.

6.5.1 Source Term Assumptions

Recommended assumptions regarding core inventory and the release of radionuclides from the fuel are in Section 3 of this appendix. The release from the breached fuel is based on Section 3.2 of this appendix and the estimate of the number of fuel rods breached.

20 Facilities licensed with, or applying for, alternate repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," USNRC, December 1998 (Ref. 18), for acceptable assumptions and methodologies for performing radiological analyses.

6.5.2 Coolant Activity Assumptions

If no or minimal[21] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification. Two cases of iodine spiking should be assumed consistent with the current licensing basis of the facility.

6.5.2.1 A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum full power value (typically 60 µCi/gm DE I-131) permitted by the technical specifications (i.e., a pre-accident iodine spike case).

6.5.2.2 The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater [K] than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 µCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. [L] Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.

6.5.3 Reactor Coolant Mixing Assumptions

The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously throughout the primary coolant.

6.5.4 Transport, Reduction and Release Assumptions

Recommended assumptions related to the transport, reduction and release of radioactive material to the environment are as follows:

6.5.4.1 The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between the affected and unaffected steam generators in such a manner that the calculated dose is maximized.

6.5.4.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lb_m/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on room

21 Minimal fuel damage is defined as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining the dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

temperature liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lb_m/ft^3).

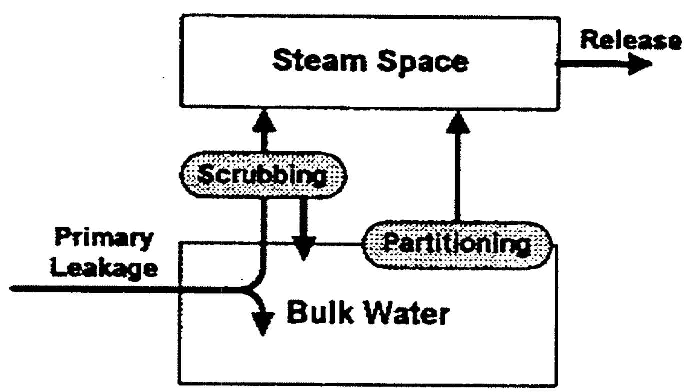
6.5.4.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

6.5.4.4 The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of off-site power when this assumption is part of the licensing basis of the facility.

6.5.4.5 The transport model used for iodine releases from the steam generators is shown in Figure 6.1 and is summarized below:

Figure 6.1

Transport Model



- A portion of the primary to secondary leakage will flash to vapor, based on the thermodynamic conditions in the primary and secondary coolant. With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence.

- The primary to secondary leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident", during periods of total submergence of the tubes.

- The primary to secondary leakage that does not immediately flash is assumed to mix with the bulk water.

- The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient.[22] A partition coefficient of 100 for iodine may be assumed.

6.5.4.6 Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip. The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.

6.5.4.7 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

6.6 Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident

This section provides assumptions for evaluating the radiological consequences of a main steam line break accident at PWR light water reactors.[23] These assumptions supplement the guidance provided in Sections 3 through 5 of this appendix.

6.6.1 Source Term Assumptions

Recommended assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Section 3 of this appendix. The release from the breached fuel is based on Section 3.2 and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.

6.6.2 Coolant Activity Assumptions

If no or minimal[24] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two

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22 The *Partition Coefficient* is defined as:

PC = *(mass of I_2 per unit mass of liquid) /(mass of I_2 per unit mass of gas)*

23 Facilities licensed with, or applying for, ARC should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," (USNRC, December 1998 (Ref. 18), for acceptable assumptions and methodologies for performing radiological analyses.

24 Minimal fuel damage is defined as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining the dose equivalent

cases of iodine spiking should be assumed consistent with the current licensing basis of the facility.

6.6.2.1 A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum full power value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a pre-accident iodine spike case).

6.6.2.2 The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins. L

6.6.3 Reactor Coolant Mixing Assumptions

The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.

6.6.4 Chemical Forms Assumptions

The chemical form of radioiodine released from the fuel should be assumed to be 5 percent particulate, 91 percent elemental and 4 percent organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

6.6.5 Transport, Reduction and Release Assumptions[25]

Recommended assumptions related to the transport, reduction and release of radioactive material to the environment are as follows.

6.6.5.1 For facilities that have not implemented alternate repair criteria (ARC), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional generator specifications (both per generator and total of all generators),

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I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

25 *Faulted* refers to the state of the steam generator in which the secondary side has been depressurized by a MSLB such that protective system response (main steam line isolation, reactor trip, safety injection, etc.) has occurred. This is also referred to as the *affected* steam generator.

the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.

6.6.5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lb_m/ft^3).

6.6.5.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

6.6.5.4 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

6.6.5.5 The transport model described in Section 6.5.4 should be used for iodine and particulates. During periods of dryout in the faulted steam generator, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation.

6.7 Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident

This section provides assumptions for evaluating the radiological consequences of a locked rotor accident at PWR light water reactors.[26] These assumptions supplement the guidance provided in Sections 3 through 5 of this appendix.

6.7.1 Source Term Assumptions

6.7.1.1 Recommended assumptions regarding core inventory and the release of radionuclides from the fuel are in Section 3 of this appendix. The release from the breached fuel is based on Section 3.2 and the estimate of the number of fuel rods breached.

6.7.1.2 If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment.

26 Facilities licensed with, or applying for, ARC should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," USNRC, December 1998 (Ref. 18), for acceptable assumptions and methodologies for performing radiological analyses.

6.7.1.3 The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.

6.7.1.4 The chemical form of radioiodine released should be assumed to be 5 percent particulate, 91 percent elemental and 4 percent organic.

6.7.2 Transport, Reduction and Release Assumptions

Recommended assumptions related to the transport, reduction and release of radioactive material to the environment are as follows.

6.7.2.1 The primary-to-secondary leak rate in the steam generators should be assumed to be the leak-rate-limiting condition for operation specified in the technical specifications. The leakage should be apportioned between the steam generators in such a manner that the calculated dose is maximized.

6.7.2.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on room temperature liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lb_m/ft^3).

6.7.2.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

6.7.2.4 The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of off-site power when this assumption is part of the licensing basis of the facility.

6.7.2.5 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

6.7.2.6 The transport model described in Section 6.5.4 should be utilized for iodine.

6.8 Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident

This section provides assumptions for evaluating the radiological consequences of a rod ejection accident at PWR light water reactors.[27] These assumptions supplement the guidance provided in Sections 3 through 5 of this appendix.

6.8.1 Source Term Assumptions

6.8.1.1 Recommended assumptions regarding core inventory are in Section 3 of this appendix. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10 percent of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100 percent of the noble gases and 25 percent of the iodines contained in that fraction are available for release from containment.[28] For the secondary system release pathway, 100 percent of the noble gases and 50 percent of the iodines in the fuel melt fraction are released to the reactor coolant.

6.8.1.2 If no or minimal[29] fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the LOCA, main steam line break, and steam generator tube rupture.

6.8.1.3 Two release cases are to be considered. In the first, 100 percent of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100 percent of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.

6.8.1.4 The chemical form of radioiodine released to the containment atmosphere should be assumed to be 5 percent particulate, 91 percent elemental and 4 percent organic. If containment sprays do not actuate or are terminated prior to accumulating sump water, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case basis. Evaluations of pH should consider the effect of acids created during the rod ejection accident event, e.g., pyrolysis and radiolysis products.

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27 Facilities licensed with, or applying for, ARC should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," USNRC, December 1998 (Ref. 17), for acceptable assumptions and methodologies for performing radiological analyses.

28 Note that Regulatory Guide 1.77 may be misleading when it states that 25 percent of the radioactive iodine inventory contained in that fraction is available for release from containment. This applies to the use of an instantaneous plateout assumption. When plateout removal is modeled mechanistically, the initial value should be 50 percent rather than 25 percent.

29 Minimal fuel damage is defined as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits.

6.8.2 Transport, Reduction and Release from Containment

Recommended assumptions related to the transport, reduction and release of radioactive material in and from the containment are as follows.

6.8.2.1 A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments or other engineered safety features may be taken into account. Refer to Section 6.1 of this appendix for guidance on acceptable methods and assumptions for evaluating these mechanisms.

6.8.2.2 The containment should be assumed to leak at the leak rate incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50 percent of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing. Leakage from subatmospheric containments is assumed to be terminated when the containment is brought to a subatmospheric condition as defined in technical specifications.

6.8.3 Transport, Reduction and Release from the Secondary System

Recommended assumptions related to the transport, reduction and release of radioactive material in and from the secondary system are as follows.

6.8.3.1 A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.

6.8.3.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lb_m/ft^3).

6.8.3.3 All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.

6.8.3.4 The transport model described in Section 6.5.4 of this appendix should be utilized for iodine.

7. REFERENCES

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8. APPENDIX C ENDNOTES

The structure of this document follows the format and guidance content of Regulatory Guide 1.183 wherever practical and applicable. This approach was used to capture the results of the interactions of the CRH Task Force and the NRC staff in a form that takes advantage of work performed recently by the NRC. Regulatory Guide 1.183 has already benefited from substantial industry review and comment. In addition, much of the guidance developed in Regulatory Guide 1.183 is applicable to the analyses described here. The common platform will facilitate communication and documentation of future developments in dose analysis assumptions, methods and applications. To retain this format to the greatest degree, these endnotes are structured to facilitate additional industry comments that provide important clarification and guidance.

A. The following approaches from Regulatory Guide 1.183 represent a relaxation from current regulatory guidance as applied to analyses with TID-14844 source terms. They were identified by examining performance features that are not a function of the timing or chemical assumptions of the alternate source term, and therefore are applicable to the traditional TID-14844 source term. This process was established as a result of the CRH Task Force discussions with NRC staff. The sections of Appendix C that describe implementation of these approaches are shown in parentheses.

1. Non-LOCA gap fractions consistent with those allowed in RG 1.183. (Section 3.2: Table 2, including footnote)

2. Consideration of enthalpy deposition methodology for assessing the amount of fuel damage from non-LOCA design basis events. (Section 3.6)

3. The assumption of a gross failure of a passive component (leading to a 50 gpm leak for one-half hour beginning at 24 hours following the start of the accident) was previously required for those plants with potential leak points not served by safety-related filtered exhaust. This assumption, which is described in Appendix B of SRP Section 15.6.5, is no longer required. Such a failure is extremely unlikely and the control room dose consequences would be expected to be low independent of safety-related ventilation. However, an assumption that should be made is that the ESF leakage used in the radiological analysis will be twice that which, as specified in the Technical Specifications or in licensee commitments to item III.D.1.1 of NUREG-0737, would require declaring such systems inoperable. (Section 6.1.4.2 and Endnote G)

4. The assumed BWR primary containment leak rate can be reduced after the first 24 hours following a LOCA to a value not less than 50 percent of the technical specification leak rate if supported by plant configuration and analysis. (Section 6.2.1.7)

5. An iodine decontamination factor of 200 allowed for 23 feet of water in a fuel handling accident. (Section 6.2.2)

6. An iodine spiking factor of 335 has been justified for application for steam generator tube rupture in comparison to the previous regulatory guidance of 500. (Section 6.5.2.2)

7. The iodine spiking duration of 8 hours for SGTR and main steam line break has been prescribed, along with the possibility of justifying a shorter duration. (Section 6.5.2.2 and Endnote J, and Section 6.6.2.2)

8. Use of ARCON96 computer code for determining control room χ/Q values. (Section 5.2 and Appendix D)

Additionally this appendix recommends the use of ICRP-30 Dose Conversion Factors (Sections 4.1.1 and 4.2.7) and an increase in the thyroid dose limit for Control Room operators from 30 Rem to 50 Rem. (Section 4.3: Endnote D in Table 4)

B. Although probabilistic risk assessments (PRAs) can provide useful insights into system performance and suggest changes in how the desired depth is achieved, defense in depth continues to be an effective way to account for uncertainties in equipment and human performance. The NRC's policy statement on the use of PRA methods calls for the use of PRA technology in all regulatory matters in a manner that complements the NRCs deterministic approach and supports the traditional defense-in-depth philosophy.

C. Licensees using or proposing to use ICRP-30 DCFs in accident calculations should determine whether this will require a revision of the facility's technical specification definition for dose equivalent I-131 to reflect these new DCFs.

D. If thyroid dose is used as the dose measure, then 50 rem or greater thyroid is more closely equivalent to five rem whole body than is the 30 rem thyroid from the SRP. General Design Criterion 19 requires that "Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident." SRP Section 6.4 establishes a Guideline of 30 rem to the thyroid as meeting the requirements of GDC 19. This interpretation has been present since the 1975 version of the SRP, NUREG-75/087. This guideline is consistent with the maximum permissible annual doses recommended in paragraph 56 of ICRP Publication 9 [1966] to the skin, thyroid and bone of 30 rem. It's also consistent with the permissible occupational dose rate of 0.6 rem/week established on page 19 of ICRP Publication 2 [1959], which equates to an annual dose limit of 0.6 Rem/week * 50 weeks. 10CFR20 has since been revised to implement the guidance of ICRP Publication 30 [1978] for annual limits on intake: "Annual limit on intake (ALI) means the derived limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year. ALI is the smaller value of intake of a given radionuclide in a year by the reference that would result in a committed effective dose equivalent of 5 rem (0.05 Sv) or a committed dose equivalent of 50 rem (0.5 Sv) to any individual organ or tissue." Based on this, a guideline value of 50 rem to the thyroid is supported as equivalent to the 5 rem whole body limit of GDC 19.

E. The bases for using containment mixing rates determined by the cooldown rate in the sprayed region and the buoyancy-driven flow that results are under current review by the NRC staff. Based on work performed by the industry to validate and demonstrate the performance of these models, improvements to containment mixing rates on the order of three to five turnovers of the unsprayed region per hour may by achievable for some system configurations and conditions. The industry position was presented to the NRC team in September 2000. Review questions highlighted the desire for additional benchmarking experience for the approach and methods. Industry responses are being reviewed by the NRC staff at the time of publication of this document.

F. The treatment of the evolution of molecular iodine vapor into the containment atmosphere in cases where the suppression pool solution maintains a pH of less than 7 is under current review by the NRC staff.

G. The assumption of a gross failure of a passive component (leading to a 50 gpm leak for one-half hour beginning at 24 hours following the start of the accident) previously required in Appendix B of SRP Section 15.6.5 for those plants with potential leak points not being served by safety-related filtered exhaust is no longer required. Such a failure is extremely unlikely and the control room dose consequences would be expected to be low independent of safety-related ventilation. However, an assumption that should be made is that the ESF leakage used in the radiological analysis will be twice that specified in the Technical Specifications or licensee commitments to item III.D.1.1 of NUREG-0737 and would require declaring such systems inoperable.

H. Regulatory Guide 1.25, Safety Guide 25, prescribes that the fuel handling accident inventory calculation should include an appropriate radial peaking factor. Minimum acceptable radial peaking factors given in Regulatory Position C.1.e of this Safety Guide are 1.5 for BWRs and 1.65 for PWRs.

I. This has been established based on CRH Task Force discussions with NRC staff.

J. For internal fuel pin pressures greater than 1200 psig, it is necessary to provide justification for the DF value that is assumed. The original database that was used to determine the DF performance needs to be examined to provide justification.

K. This has been established based on CRH Task Force discussions with NRC staff.

L. Based on CRH Task Force discussions with NRC staff, 8 hours has been determined as the maximum time of duration of the spike. The CRH Task Force has provided additional justification for a reduction in these release times. This issue is under NRC review.

APPENDIX D

ATMOSPHERIC DISPERSION

1. PURPOSE/SCOPE

This appendix provides guidance for performing atmospheric dispersion calculations in support of control room habitability analyses for design basis radiological accidents as described in Appendix C. Note that guidance on atmospheric dispersion analyses contained in this section applies only to radiological analyses; guidance for performing atmospheric dispersion calculations in support of toxic gas assessments is contained in Appendix G.

2. PERFORMING ATMOSPHERIC DISPERSION ANALYSES

A major factor in most control room radiological analyses is the dispersion of the radioactive plume and the resulting concentration at the control room intake. Various studies have been made over the last 50 years to quantify atmospheric dispersion. The most widely used empirical fit to field data is a model based on plume dispersion with a Gaussian distribution of pollutants in all three dimensions.

Atmospheric dispersion factors (also known as relative concentration or χ/Q values) are generally difficult to determine when both the release point and the receptor are located within or near atmospheric turbulence created by a complex of buildings. Several attempts have been made to overcome this shortcoming since an accurate prediction of χ/Q values for short distances in the vicinity of buildings is needed for control room habitability analyses.

Licensees have the option to use the methodology currently in their licensing basis in lieu of the methodologies presented in this appendix.

2.1 MURPHY-CAMPE

2.1.1 BACKGROUND

The Murphy-Campe methodology was first proposed by U.S. Atomic Energy Commission (AEC) staff members K. G. Murphy and K. M. Campe in a paper presented at the 13th AEC Air Cleaning Conference (Reference 1). It was later cited in Section 6.4 of the Standard Review Plan (NUREG-0800) as the appropriate method for evaluating atmospheric dispersion for the design basis accidents. The Murphy-Campe methodology has been in use since the mid-1970s and is the standard for most modern plants.

Murphy and Campe based their methodology on a number of wind tunnel and field tests that had been performed on specific building configurations that were available when they wrote their paper. Although these wind tunnel and field tests had resulted in usable information for specific situations, Murphy and Campe acknowledged that general applicability was not assured. However, to provide a basis for evaluations, the AEC staff formulated an interim position using conservative interpretations of the available data. Thus, the Murphy-Campe methodology was intended to be a bounding-type calculation requiring little site-specific information.

2.1.2 Implementation

The Murphy-Campe methodology consists of first determining the five percentile χ/Q value (defined as the χ/Q value that is exceeded not more than 5 percent of the time at the specific site in question) which is used as the χ/Q value for the first post-accident time interval. Meteorological conditions typically associated with the Murphy-Campe five percentile χ/Q values are F stability with wind speeds around one meter/sec. The determination of χ/Q values for subsequent time intervals involves corrections for long-term meteorological averaging for wind speed and wind direction.

To perform calculations using the Murphy-Campe methodology, refer to SRP 6.4 and its reference to the Murphy-Campe paper presented at the 13th AEC Air Cleaning Conference.

2.2 ARCON96

2.2.1 Background

By the mid-1980s, after a number of atmospheric dispersion field tests were conducted within building complexes, it became apparent the Murphy-Campe methodology tended to overestimate relative concentrations during low wind speed conditions. Consequently, the development of a more robust methodology that would better describe atmospheric dispersion near buildings became desirable. The Pacific Northwest National Laboratory under contract with the NRC subsequently developed ARCON96 (Reference 2). The model uses detailed meteorological field data and, as such, shows improved performance in predicting the effect of building wakes, particularly under light wind conditions.

2.2.2 Description

Per NUREG/CR-6331, the basic diffusion model implemented in the ARCON96 code is a straight-line Gaussian model that assumes the release rate is constant for the entire period of release. This assumption is made to permit evaluation of potential effects of accidental releases without having to specify a complete release sequence.

ARCON96 permits evaluation of ground level, vent and elevated releases. Building wake effects are considered in the evaluation of relative concentrations from ground level releases. Vent releases are treated as a mixed mode (part time ground level, part time elevated) release where the proportion of the ground-vs.-elevated mixture is determined by the ratio between the effluent vertical velocity and the release-height wind speed. Elevated releases are corrected for stack downwash and differences in terrain elevation between the stack and the receptor (e.g., control room intake).

Diffusion coefficients used in ARCON96 have three components. The first component is the diffusion coefficient used in other NRC models such as XOQDOQ (Reference 3) and PAVAN (Reference 4). The other two components are corrections to account for enhanced dispersion under low wind speed conditions and in building wakes.

Derivations of the low wind speed and building wake corrections are based on analysis of diffusion data collected in various building wake diffusion experiments (Reference 5). The experiments were conducted under a wide range of meteorological conditions. The wake correction model included in ARCON96 treats diffusion under low wind speed conditions with improved results.

ARCON96 calculates relative concentrations using hourly meteorological data. The resulting hourly averages are then combined to estimate concentrations for periods ranging in duration from 2 hours to 30 days. Wind direction is considered as the averages are formed. As a result, the averages account for persistence in both diffusion conditions and wind direction. Cumulative frequency distributions are prepared from the average relative concentrations. Relative concentrations that are exceeded no more than 5 percent of the time are determined from the cumulative frequency distributions for each averaging period. Finally, the relative concentrations for 5 standard averaging periods used in control room habitability assessments are calculated from the 5th percentile relative concentrations.

Although ARCON96 is based on a simple Gaussian dispersion model, the χ/Q values predicted by the model do not vary inversely with the wind speed for all wind speeds because the building wake correction is not a linear function of wind speed. In addition, the building wake corrections are not particularly sensitive to atmospheric stability. Consequently, unlike the Murphy-Campe methodology, F stability and a wind speed of 1 meter/sec do not generate the five percentile χ/Q values for ground level releases and receptors. The ARCON96 five percentile χ/Q values are typically associated with wind speeds of 3 to 4 meter/sec.

2.2.3 Implementation

The May 9, 1997, version of the ARCON96 computer code as described in Revision 1 to NUREG/CR-6331 is an acceptable methodology for assessing control room χ/Q values for use in design basis accident radiological analyses, unless unusual siting, building arrangement, release characterization, source-receptor configuration, meteorological regimes or terrain conditions indicate otherwise. Use of the ARCON96 computer code is subject to the conditions discussed below. In addition, Table D-1 identifies each ARCON96 input and acceptable values (or range of values) for each input.

a. Software QA Program

The ARCON96 code should be obtained and maintained under an appropriate software quality assurance program that complies with the applicable criteria of 10CFR50 Appendix B and meets other applicable industry consensus standards. Each licensee is ultimately responsible for the accuracy and appropriateness of use of the ARCON96 results.

b. Meteorological Data Base

Meteorological data input to ARCON96 should be obtained from instrumentation that is maintained under the site's meteorological measurements program as described in the facility's licensing basis. Three to five years of hourly observations should be used. If fewer data are used, additional evaluations may be necessary to demonstrate that the lesser data period used is representative of long-term meteorological trends at the site.

c. Receptor Location Selection

All potential locations from which the control room may draw air from the environment must be considered as an intake. This includes all ventilation system intakes and infiltration locations, such as doors and penetrations. The potential intakes may change over the course of the accident due to plant system response or manual operator actions. The system assessment outlined in Appendix H can be used to identify the potential locations of significant infiltration.

- A χ/Q value should be evaluated for each release-intake combination. It may be possible to qualitatively show that the χ/Q values for some release-intake combinations would be bounded by values calculated for other combinations and thereby reduce the number of needed calculations.
- The most restrictive (highest) χ/Q value for each release-intake combination should be used.
- For control rooms with dual intake designs, the guidance of Section III.D and Figure 1 of the Murphy-Campe paper should be applied. In addition, the practice of determining the χ/Q value for the more restrictive intake and dividing by two is acceptable only if it can be shown that the two intakes have equal flow rates and are not simultaneously within the wind direction window for any given wind direction.

d. Release Types

ARCON96 models three different release types: ground level, vent and stack.

i. Ground Level Release

The ground level release type is appropriate for the majority of control room atmospheric dispersion assessments. The default ground level source is typically treated as a point-source formulation. However, in some situations, ground level releases can be better classified as area sources. Examples might include postulated releases from the surface of a reactor or secondary containment building or releases from multiple points such as the roof vents on typical turbine buildings. ARCON96 reduces an area source to a virtual point source using two initial diffusion coefficients provided by the code user.

- LOCA radiological analyses have typically assumed that the containment structure could leak anywhere on the exposed surface. As such, these analyses typically use the shortest distance between the containment surface and the control room intake and treats the containment as a point source. This approach may have been unnecessarily conservative. A more reasonable approach is to model the containment surface as a vertical area source with ARCON96. This treatment is acceptable for design basis calculations provided that it is used in conjunction with the total release rate (e.g., Ci/sec) from the containment.
- Since leakage is more likely to occur at a penetration, the potential impact of containment penetrations exposed to the environment within the modeled area should be considered. It may be necessary to consider several cases to ensure that the χ/Q value for the most limited location is assigned. Penetrations that are exposed within safety-related structures need not be considered.
- The height and width of the area source (e.g., the containment surface) should be the maximum vertical and horizontal dimensions of the building cross-section area perpendicular to the line of sight to the control room intake. In the absence of site-specific empirical data, the initial diffusion coefficients are found by:

$$\sigma_y = (\text{area source width})/6$$

$$\sigma_z = (\text{area source height})/6$$

The shortest horizontal distance from the building surface to the receptor should be used as the source-receptor distance. The release height should be set on the surface of the area source that will result in the shortest slant path.

- Multiple roof vents can be modeled as a rectangular area source if the assumed rectangle encompasses all the vents and the release rate from each vent is approximately the same. The distance to the receptor is measured from the closest point on the assumed rectangular source and, in the absence of site-specific empirical data, the initial diffusion coefficients are found by:

  σ_y = (length of the nearest side of the rectangular area source)/6

  $\sigma_z = 0.0$

  If vents are clustered such that a rectangular area cannot adequately represent the postulated effluent release distribution, then another approximation methodology should be used.

ii. Vent Release

The vent release type was intended for use with uncapped upward-directed vents on or slightly above building surfaces. This model is considered appropriate for use in long-term routine effluent calculations but is considered inappropriate for the short-term releases associated with accident assessments. As such, the vent release type should not be used in design basis accident applications.

iii. Stack Release

The stack release type is appropriate for releases from standalone stacks that are 2.5 times the height of adjacent solid structures, or as established in the plant's licensing/design basis. Otherwise, the release should be considered a ground level release. At this time the NRC does not allow the use of plume rise from buoyancy and mechanical jet effects in determining an effective stack height. Use of the elevated plume option may lead to unrealistically low concentrations at control room intakes located close to the base of tall stacks. If the χ/Q values calculated in this situation are all extremely low, other models should be used to estimate the potential control room intake χ/Q values.[30]

Fumigation conditions should be considered using the guidance of regulatory positions 1.3.2.b, 2.1.2, and 2.2.2 of Regulatory Guide 1.145.[31] Ground level χ/Q values generated by ARCON96 may be substituted for values generated with Equation 5 of Regulatory Guide 1.145.

30 At the time this appendix was being written, consideration was being given to upgrading ARCON96's handling of stack releases to allow its unconditional use in generating χ/Q values for stack releases.
31 For facilities that are implementing or have implemented an alternative source term, fumigation conditions should be assumed to exist at the onset of the major radioactivity releases in lieu of the start of the accident as specified in Regulatory Guide 1.145.

3. Site Specific Diffusion Tests

Appropriately structured site-specific atmospheric diffusion tests can be considered as an alternative to the analytical methods presented above. Such tests (e.g., field, wind tunnel) must encompass a sufficient range of meteorological and, if appropriate, modeling conditions applicable to the site so as to ensure that the limiting case(s) have been evaluated. The testing and results obtained should be verified and validated.

At this time the NRC staff has not accepted the results from site-specific wind tunnel tests for use in licensing analyses. Direct comparisons between wind tunnel test results and applicable field measurements may facilitate staff acceptance.

4. References

1. K.G. Murphy and K.W. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," published in *Proceedings of 13th AEC Air Cleaning Conference, San Francisco, CA,* CONF-740807, U.S. Atomic Energy Commission (now USNRC), August 1974.

2. J.V. Ramsdell, Jr. and C.A. Simonen, "Atmospheric Relative Concentrations in Building Wakes," NUREG-6331, Revision 1, U.S. Nuclear Regulatory Commission, May 1997.

3. USNRC, "XOQDOQ: Computer Program for the Meteorological Evaluation of Routine Releases at Nuclear Power Stations," NUREG/CR-2919, U.S. Nuclear Regulatory Commission, 1982.

4. T.J. Bander, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," NUREG/CR-2858, U.S. Nuclear Regulatory Commission, November 1982.

5. J.V. Ramsdell, Jr. and C.J. Fosmire, "Atmospheric Dispersion Estimates in the Vicinity of Buildings," PNL-10286, Pacific Northwest Laboratory, Richland, Washington, 1995.

6. USNRC, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, Revision 1, U.S. Nuclear Regulatory Commission, Reissued February 1983.

Table D-1
ARCON96 Input Parameter Guidance

Parameter Type	Parameter	Discussion	Acceptable Input
Meteorological Data	Lower Measurement Height (meters)	This value is used to adjust wind speeds for differences between the heights of the instrumentation and the release.	Use the actual instrumentation height when known; otherwise, assume 10 meters. This value may not exceed 100 m.
	Upper Measurement Height (meters)	This value is used to adjust wind speeds for differences between the heights of the instrumentation and the release.	Use the actual instrumentation height; otherwise, use the height of the containment or the stack height, as appropriate. If wind speed measurements are available at more than two elevations, the instrumentation at the height closest to the release height should be used. This value may not exceed 300 m.
	Wind Speed Units	Wind speed can be entered as either miles per hour, meters per second or knots.	Use the wind speed units that correspond to the units of the wind speeds in the meteorological data file.
Receptor Data	Distance to Receptor (meters)	This value is used for calculating the slant range distance for ground level releases and the off-centerline correction factors for stack releases.	Use the actual straight-line horizontal distance between the release point and the receptor of interest (e.g., control room intake). For ground level releases, it may be appropriate to consider flow around an intervening building if the building is sufficiently tall that it is unrealistic to expect flow from the release point to go over the building. This distance may not exceed 10,000 m. NOTE: If the distance to the receptor is less than about 10 meters, ARCON96 should not be used to derive relative concentrations.
	Intake Height (meters)	This value is used for calculating the slant range distance for ground level releases and the off-centerline correction factors for stack releases.	Use the height of the intake above ground level. If the intake height is not available for ground level releases, assume the intake height is equal to the release height. This value may not exceed 100 m.

Table D-1
ARCON96 Input Parameter Guidance

Parameter Type	Parameter	Discussion	Acceptable Input
	Elevation Difference (meters)	This value is used to normalize the release heights and the receptor heights in those cases where the two heights are specified as "above grade" with different grades for the release point and intake height, or where one measurement is referenced to "above grade" and the other "above sea level."	Use zero if the release point and the receptor are on the same structure or the heights of the release point and the receptor are measured from the same reference plane. If there is a difference in height between the release point and receptor reference planes, enter a positive value if the grade elevation at the release point is higher or enter a negative value if the grade elevation at the receptor is higher. This value must be between -1,000 m and +1,000 m.
	Direction to Source (degrees)	This value is used along with the wind direction window to establish the range of wind directions that should be included in the assessment of χ/Q values.	Use the direction from the receptor back to the release point. The direction entered must have the same point of reference as the wind directions reported in the meteorological data (i.e., some facilities list a "plant north" on-site arrangement drawings that differ from "true north"). For ground level releases, if the plume is assumed to flow around a building rather than over it, the direction may need to be modified to account for the redirected flow. In this case, χ/Q values should be calculated assuming flow around and flow over (through) the building and the higher of the two values should be used.
Source Data	Release Type	Releases can be identified as ground, vent or stack. Building wake effects are considered in the evaluation of ground level releases; vent releases are treated as a mix of ground level and elevated releases; and stack releases are treated as elevated releases.	All releases should be classified as either ground or stack releases; the vent release model should not be used for DBA accident calculations. Unless the actual release point is more than 2.5 times the height of structures in the vicinity of the stack, or the plant's licensing basis has already defined the release point as an elevated release, the release should be classified as ground.

Table D-1
ARCON96 Input Parameter Guidance

Parameter Type	Parameter	Discussion	Acceptable Input
	Release Height (meters)	This value is used to adjust wind speeds for differences between the heights of the instrumentation and the release, to determine the slant path distance for ground level releases and to correct off-centerline data for elevated releases.	Use the actual release height whenever available; otherwise, set the release height to the intake height. This value must be between 1 m and 300 m. Plume rise from buoyancy and mechanical jet effects may be considered in establishing the release height if it can be demonstrated with reasonable assurance that the vertical velocity of the release will be maintained during the course of the accident.[32]
	Building Area (m^2)	This value is used in the high wind speed adjustment for the ground level and vent release models.	Use the actual building vertical cross-sectional area perpendicular to the wind direction whenever available; otherwise, use a default value of 2000 m^2. This value must be between 0.01 m^2 and 10,000 m^2. This building area is for the building(s) that have the largest impact on the building wake within the wind direction window (usually the reactor containment). Note that, for diffuse sources, the building area entered here may be different from that used to establish the diffuse source.
	Vertical Exit Velocity (m/sec)	This value is used for determining the amount of the plume that enters the building wake for the vent release model and for stack downwash calculations for the vent and stack release models.	Use a value of zero unless it can be demonstrated with reasonable assurance that the value will be maintained during the course of the accident. If the vent or stack is capped, use a value of zero. This value must be between 0 m/sec and 50 m/sec.

32 At the time this appendix was being written, consideration was being given to upgrading ARCON96 to perform plume rise calculations for high energy releases such as atmospheric dump valve, power-operated relief valve, or main steam safety valve discharges associated with a steam generator tube rupture DBA.

Table D-1
ARCON96 Input Parameter Guidance

Parameter Type	Parameter	Discussion	Acceptable Input
	Stack Flow (m^3/sec)	This value is used for all release models to ensure that the near field concentration is not greater than the concentration at the release point.	Use a value of zero unless it can be demonstrated with reasonable assurance that the value will be maintained during the course of the accident. This value must be between 0 m^3/sec and 100 m^3/sec.[33]
	Stack Radius (meters)	This value is used for the vent and stack release models to determine the maximum stack height reduction during downwash conditions.	If the vertical velocity entered is not zero, use the actual stack internal radius; otherwise, use a value of zero. This value must be between 0 m and 10 m.
Default Data	Surface Roughness Length (meters)	This value is used in adjusting wind speeds to account for differences in meteorological instrumentation height and release height.	In lieu of the default value of 0.1, use a value of 0.2 for most sites. Valid values range from 0.1 for sites with low surface vegetation to 0.5 for forest covered sites.
	Wind Direction Window (degrees)	This value is used along with the direction to source to establish the range of wind directions to be included in the assessment of χ/Q values.	Use the default value of 90 degrees (which represents a range of wind direction 45 degrees on either side of line of sight from the source to the receptor).
	Minimum Wind Speed (m/sec)	This value is used to identify calm conditions.	Use the default wind speed value of 0.5 m/sec (regardless of the wind speed units entered earlier) unless there is some indication that the anemometer threshold is greater than 0.6 m/sec.
	Averaging Sector Width Constant	This value is used to prevent inconsistency between the centerline and sector-average χ/Q values for wide plumes (has largest effect on ground level plumes).	Use a value of 4.3 in lieu of the default value of 4.0.

33 The 100 m^3/sec stack flow limit is an arbitrary upper range check value programmed into the graphical program interface. If justifiable, the user can execute ARCON96 with higher stack flow values by editing the input file generated by the graphical program interface with a text editor and executing the code in batch mode.

Table D-1
ARCON96 Input Parameter Guidance

Parameter Type	Parameter	Discussion	Acceptable Input
	Initial Diffusion Coefficients (meters)	These values define the initial diffusion coefficients that define area sources.	For an area source such as a containment surface, in the absence of site-specific empirical data, use: σ_y = (area source width)/6 σz = (area source height)/6 For an area source such as multiple roof vents that can be modeled as a rectangular area source, in the absence of site-specific empirical data, use: σ_y = (length of the nearest side of the rectangular area source)/6 $\sigma_z = 0.0$ Otherwise, use $\sigma_y=\sigma_z=0$.
	Hours in Averages	These values specify the number of hours for each averaging period.	Use the default values.
	Minimum Number of Hours	These values specify the minimum number of hours for a valid average.	Use the default values.

APPENDIX E

SMOKE INFILTRATION IMPACT ON SAFE SHUTDOWN

1. PURPOSE/SCOPE

This appendix provides a qualitative assessment tool for managing the issue of smoke infiltration as described in Section 6. The guidance ensures that the operator maintains an ability to safely shut down the plant during a fire/smoke event originating outside the control room.

2. ASSESSMENT

Perform an assessment to assure that the operator has the capability to safely shut down the plant from either the control room or the remote shutdown locations during a single credible fire/smoke event. The following items should be considered:

- Verify that the remote shutdown panel or controls are not located within the control room habitability envelope.
- Verify that the remote shutdown panels or controls and the control room are adequately separated by distance, or appropriate fire barriers, such that a single credible fire/smoke event in one area could not affect the habitability of the other.
- Verify that a credible fire/smoke event does not exist that could affect control room habitability while simultaneously blocking the normal egress path to the remote shutdown panels or controls. If not, verify that an alternate egress path exists and that it is addressed in plant procedures. Although desirable, this guidance does not require that the alternate route be equipped with emergency lighting to specifically cover this scenario.
- Verify that sufficient procedural guidance exists to mitigate credible fire/smoke events. Fire/smoke response procedures should contain provisions to manually align ventilation systems to exhaust smoke away from the control room when practical.
- Verify that a sufficient number of control room operators per shift are qualified in the use self-contained breathing apparatus (SCBA) to safely shut-down the plant. Certain success paths to achieve the stated goal above may require the limited use of SCBA.
- Verify that the appropriate SCBA and smoke removal equipment are available and properly staged.
- Verify that initial and continuing training is performed to ensure familiarity with the success paths discussed in this appendix.

3. SUCCESS PATH LOGIC

The steps below outline possible success paths to ensure safe shutdown capability is maintained during a smoke infiltration event. These paths should provide confidence that a serious smoke infiltration event can be mitigated.

- Should an excessive amount of smoke infiltrate the control room envelope, the operators may isolate the ventilation system if the outside air intake is the primary entry point of the smoke. Efforts should then be taken to clear the smoke using either an installed smoke removal system or portable blowers. A short-term limited use of SCBAs may be expected in this situation. The ability to clear the smoke in a reasonable period of time would be considered a success path.
- If smoke removal is not a success path in the short term, then assess if the smoke is having a detrimental effect on the operator's ability to control the plant. Consideration should be given to evacuate to the remote shutdown panels or controls. This decision would be based on the severity of the situation and the availability of a safe egress path to the remote shutdown panels.
- If the remote shutdown panels or controls are also contaminated with smoke, it may be advantageous to remain in the control Room using SCBAs until smoke can be cleared from one of the locations.
- If the decision is made to evacuate the control room, choose a primary or an alternate path to the remote shutdown panels or controls that is least affected by the event. It may be necessary to use SCBA while transiting to the remote shutdown panels or controls.
- If the assessment determines that a potential situation exists where a success path is not assured, the condition should be entered into the plant's corrective action process to ensure an appropriate resolution.

APPENDIX F

COMPENSATORY MEASURES ALLOWABLE ON AN INTERIM BASIS

1. PURPOSE/SCOPE

Licensees may need to implement compensatory measures as part of the plant corrective action program. This appendix identifies two actions that may be considered for use as compensatory measures in the event of unacceptable radiological release consequences. These actions are the use of self-contained breathing apparatus (SCBA) and the use of potassium iodide (KI) tablets. Other plant specific compensatory actions may be appropriate. The use of any compensatory measure will require a plant specific evaluation to justify its use.

The use of SCBA and KI has been determined to be acceptable for addressing control room envelope integrity in the interim situation until the licensee remediates the issue. However, use of SCBA or KI in the mitigation of situations where in-leakage does not meet design basis limits is not acceptable as a permanent solution. 10CFR20.1701 essentially says that engineering/process controls shall be used to the extent practical. If not practical, then 10CFR20.1702 methods should be used. Therefore, the use of SCBAs should be a last resort. The length of time for which credit is allowable should be determined on a case-by-case basis. If credit is currently part of the licensing basis, special considerations may be necessary.

2. COMPENSATORY MEASURES

In addition to the compensatory measures addressed in this appendix, plant modifications such as the installation of local toxic gas monitors should be considered in the event of unacceptable toxic gas release consequences. Additional guidance is provided in Draft Regulatory Guide DG-1087, *Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release.*

Credit for SCBA as a compensatory measure is allowed provided an approved respiratory protection program is in effect. The following are key considerations for crediting SCBA use in support of control room habitability assessments.

2.1 An Approved Respiratory Protection Program Should be in Effect

a) An approved respiratory protection program in accordance with 10 CFR Part 20, Appendix H, Regulatory Guide 8.15, Rev. 1, *Acceptable Programs for Respiratory Protection* and NUREG-0041, Rev. 1, *Manual of Respiration, Protection Against Airborne Radioactive Materials* is established and in place.

i. Maintaining an adequate respiratory protection program is vital to workers' safety and, thus, to their ability to respond in a timely fashion to emergencies.

ii. Plant operators and emergency response workers can face not only radiological airborne hazards, but, in many cases, are challenged by unknown and potentially immediately dangerous to life and health (IDLH) conditions. Therefore, non-radiological hazards should also be considered.

b) Plans for dealing with emergencies should include consideration of:

i. Postulated duration of SCBA use

ii. Quantities and kinds of materials against which protection must be provided

iii. Physical characteristics of the hazardous area

iv. Access requirements

v. Numbers of people and technical skills needed

vi. Amounts, types and locations of equipment necessary

vii. Need for and availability of backup/replacement supplies for use in emergencies

viii. Enhancement of communications

ix. Capability of control room faculties to accommodate operators operating with SCBA

x. Visual impairment

2.2 Sufficient Number of Operators Should be Trained and Qualified in SCBA Use.

a) The licensee should ensure there will always be sufficient numbers of control room operators on shift that are qualified for SCBA use.

b) Since SCBA use is expected to be infrequent, there should be adequate periodic, hands-on training and practice with donning and wearing SCBA including communication techniques and vision impairment during SCBA use.

c) Additionally, operators should be trained and practiced to change out air cylinders and know where spare charged air cylinders are stored for emergency use.

d) Effective program oversight and controls should be in place for tracking and maintaining operators' required periodic retraining and SCBA fit testing.

2.3 Adequate Supplies of Equipment Should be Available.

a) Sufficient dedicated, surveyed, and inventoried equipment with various size face pieces should be available for use by control room operators at all times.

b) A sufficient number of support personnel should be assigned to transport and replenish supplies for the duration of the need for SCBA.

2.4 Corrective Lenses (If required) Should be Available for SCBA Users.

a) In accordance with 10 CFR Part 20.1702(e), all those requiring vision correction should use contact lens or approved spectacle adapters.

b) A lack of required vision correction could hamper the control room operator's performance of licensed duties, including timely and effective response to emergencies.

c) Corrective lenses with temple bars interfering with the sealing surface of any respirator facepiece shall not be worn while using such equipment.

d) Semi-permeable prescription contact lenses may be worn if their use has been satisfactorily demonstrated.

e) Hard contact lenses should not be worn with full-facepiece respirators. Hard contact lenses present a distinct hazard to the individual due to the possibility of the lenses slipping because of pressure on the outside corners of the eye from a full face mask or a speck of dirt getting under them while the respirator is being worn.

2.5 Persons Using Tight Fitting (Facepiece) Respirators Should Not Have Any Facial That Interferes With the Sealing Surfaces of the Respirator.

2.6 The Required Minimum Staffing of Control Room Operators Qualified in SCBA Use Should be Clean-Shaven. Adequate Method(s) to Refill SCBA Air Cylinders Should be Available.

a) This includes proper location of air compressor intakes (e.g., not down-wind from release points).

b) When a compressor is used, it should be properly monitored and attended to ensure that the air intake remains in an uncontaminated atmosphere.

2.7 Provide For Adequate Relief From Respirator Use.

a) Provisions should be considered for operators wearing SCBA to leave the area if necessary.

2.8 Ensure an Appropriate Monitoring Program Exists.

a) An appropriate air sampling program should be implemented to monitor control room airborne radioactivity levels to determine individual exposure levels based on stay times, protection factors and respirator usage.

b) Protection factors apply only in a respiratory protection program that meets the requirements of 10 CFR Part 20.

   i) These protection factors are applicable to radiological hazards, oxygen deficiency, toxic gas and smoke/fire hazards and may not be appropriate for hazards that involve skin adsorption.

   ii) Prompt emergency response does not lend itself to pre-work assessment of airborne hazards. In emergency situations, it is clearly illogical to take a "no-protection" assumption for entry into IDLH areas of unknown hazards.

3. Use of Potassium Iodide as a Compensatory Measure for Control Room Thyroid Dose

Certain forms of iodine help the thyroid gland work correctly. Most people consume the iodine their thyroid needs from foods such as iodized salt and fish. However, the thyroid can hold or store only a certain amount of iodine. In the event of a nuclear accident involving the release of large amounts of radioiodines, significant uptake of radioiodines by the thyroid could occur from inhalation and ingestion. The basis for using KI to limit thyroid dose is that administration of stable iodide as a prophylaxis can prevent thyroidal uptake of radioiodines, and thus reduce radioactive dose to the thyroid post-accident.

KI is an effective thyroid blocking agent when administered immediately before or after an exposure to radioactive iodine (that is, within one to two hours). If KI is administered more than four hours after an acute inhalation or ingestion of radioiodine, then its effectiveness as thyroid blocking agent is substantially reduced. The prompt administration of KI in the event of a nuclear accident is critical to its effectiveness as a protective measure. Credit may be taken for a factor of 10 reduction in thyroid dose due to the administration of KI. Plant procedures should be in place to ensure KI can be administered to control room operators (and to oncoming shifts) soon after the start of an event where radioiodine has been released or could be released.

The recommended dose is 130 mg (one tablet) of potassium iodide, equivalent to 100 mg of iodide, taken by mouth. Higher doses are not required or beneficial. Additional daily administration may be required (i.e., 3 to 7 days after the accident if radioiodine releases continue). To take credit for KI as a protective measure for control room operator thyroid dose, the following actions should be implemented. Some of these actions may already be in place as part of the licensee's emergency plan procedures.

3.1 Key Considerations for Crediting KI Use in Support of Control Room Habitability Assessments

3.1.1 Although KI is a non-prescription medication, the licensee's internal policies on administering medications to employees should be reviewed and followed as required.

3.1.2 Personnel who are candidates for receiving KI must be screened for possible allergic reactions to iodine. Shift personnel who are allergic to KI may need to be temporarily reassigned, or provisions made for relieving them from duty in the event of a radioiodine release.

3.1.3 Personnel who are identified as candidates to receive KI after an accident must be on an approved list. The approved list should be readily accessible so that prompt administration can be performed.

3.1.4 It is not mandatory for control room operators to take KI as a protective measure. Those who choose not to take KI should evacuate the control room and be replaced by another qualified operator.

3.1.5 Adequate supplies of KI must be available in the control room for control room operators. Provisions must be made for storing KI tablets properly, and for periodic replacement prior to the shelf life being exceeded. Adequate supplies should also be available to administer KI to relief personnel.

3.1.6 Plant procedures should be in place to direct administration of KI to control room personnel within two hours of a radioiodine release. Procedures should also be in place to administer KI to on-coming shifts as necessary if radioiodine releases continue.

3.1.7 Controls should be in place to determine if follow-up administration of KI is required. The decision to have follow-up administration of KI should be done in consultation with the licensee's company medical representative and the plant's emergency response organization. If required, administration should occur within 3 to 7 days following the accident.

4. References and Supporting Information

1. USNRC, "Task III.D.3: Worker Radiation Protection Improvement (Revision 3), TMI Action Item III.D.3.2 (4), Develop Air Purifying Respirator Radioiodine Cartridge Testing and Certification Criteria," *Clarification of TMI Action Item Requirements*, NUREG-0737, U.S. Nuclear Regulatory Commission, 1980.
2. 10 CFR 20, "Respiratory Protection and Controls to Restrict Internal Exposures," Part 20 (RIN 3150-AF81), Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.
3. 10 CFR 20, Appendix A, "Assigned Protection Factors (APF) for Respirators," Part 20, Appendis A, Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.
4. USNRC, "Problems With Emergency Preparedness Respiratory Protection Programs," NRC Information Notice 98-20, U.S. Nuclear Regulatory Commission, June 3, 1998.
5. USNRC, "Acceptable Programs For Respiratory Protection," Regulatory Guide 8.15, , U.S. Nuclear Regulatory Commission, October 1976.
6. USNRC, "Manual of Respiratory Protection Against Airborne Radioactive Materials," NUREG-0041, U.S. Nuclear Regulatory Commission, October 1976.
7. USNRC, "Inadvertent Discharge Of Carbon Dioxide Fire Protection System And Gas Migration," NRC Information Notice 99-05, U.S. Nuclear Regulatory Commission, March 8, 1999.
8. USNRC, "Guidance Concerning 10 CFR 20.103 and Use of Pressure Demand SCBA's," HPPOS-094, U.S. Nuclear Regulatory Commission, 1991.
9. USNRC, "OSHA Interpretation: Beards and Tight-Fitting Respirators," HPPOS-116, U.S. Nuclear Regulatory Commission, 1991.
10. David C. Aldrich and Roger M. Blond, "Examination of the Use of Potassium Iodide (KI) as an Emergency Protective Measure for Nuclear Reactor Accidents," NUREG/CR-1433, U.S. Nuclear Regulatory Commission, 1980.
11. H. Behling, K. Behling and H. Amarasooriya, "An Analysis of Potassium Iodide (KI) Prophylaxis for the General Public in the Event of a Nuclear Accident," NUREG/CR-6310, U.S. Nuclear Regulatory Commission, 1995.
12. NCRP, "Protection of the Thyroid Gland in the Event of Releases of Radioiodine," NCRP Report No. 55, National Council on Radiation Protection and Measurements, August 1, 1977.

APPENDIX G

TOXIC GAS ASSESSMENTS

1. PURPOSE

Guidance is provided on performing an assessment of a hazardous chemical challenge to control room habitability. In addition, this appendix provides guidance for determining when a periodic reassessment of toxic gas challenges is warranted.

2. SCOPE

This appendix applies to the release of hazardous chemicals from mobile or stationary sources, located off-site or on-site. It does not consider the explosive or flammability hazards of these chemicals, which are considered beyond the scope of this appendix.

3. REGULATORY BASIS

General Design Criterion (GDC) 4, *Environmental and Missile Design Basis*, of Appendix A to 10 CFR Part 50 requires that structures, systems and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents. GDC 19, *Control Room*, requires that a control room be provided from which actions can be taken to operate the nuclear power plant safely under normal conditions and to maintain it in a safe condition under accident conditions.

4. PERFORMING THE TOXIC GAS ASSESSMENT

The control room of a nuclear power plant should be appropriately protected from hazardous chemicals that may be discharged as a result of equipment failures, operator errors or events and conditions outside the control of the nuclear power plant. Potential sources of hazardous chemicals may be mobile or stationary and include storage tanks, pipelines, fire-fighting equipment, tank trucks, railroad cars and barges.

Much of the guidance presented in this appendix was obtained from Regulatory Guide 1.78, *Assumption for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Chemical Release.*[34] This appendix also provides guidance beyond

34 Proposed revisions to Regulatory Guide 1.78 have been issued by the NRC in draft guide DG-1087. The proposed revision incorporates Regulatory Guide 1.95 (Reference 2) for protection of control room operators against accidental chlorine releases. The guidance related to chlorine releases provided in Regulatory Guide 1.95 and in DG-1087 is not presented in this appendix; the reader is encouraged to refer to the above publications for chlorine-specific concerns.

that contained in Regulatory Guide 1.78 in the areas of specifying toxicity limits, identifying sources of on-site and off-site hazardous materials, determining hazardous chemical release characteristics and applying updated atmospheric dispersion modeling techniques, including dense gas atmospheric dispersion models.[35] Licensees following the guidance of this appendix may use the:

- methodology that currently serves as their licensing basis,
- immediately dangerous to life and health (IDLH) exposure levels,
- guidance presented in Regulatory Guide 1.78 as supplemented by this appendix or
- other regulatory guidance (e.g., DG-1087) that is subsequently published by the NRC.

4.1 Identifying Hazardous Materials

4.1.1 Off-Site

Two federal laws were developed to provide information regarding hazardous chemicals at industrial facilities. The EPA and state and local governments maintain these data. Much of the information is easily available on the Internet or from state and local governments who receive reports from facilities.

The U.S. Department of Transportation Research and Special Program Administration maintains a HAZMAT database. The Emergency Planning and Community Right-to-Know Act (EPCRA) and the Clean Air Act Risk Management Program (RMP) require facilities to report on hazardous chemicals they store or handle. Both provide for public access to the information on these chemicals. The two regional government agencies that receive the information are the Local Emergency Planning Committee (LEPC) and the State Emergency Response Commission (SERC). The information available from reporting facilities includes annual chemical inventories or lists of chemicals stored or handled, and accident data like worst-case release scenarios.

It is important to remember that only certain toxic chemical releases need to be considered. The number of facilities covered, for example, may be limited because only certain chemicals and threshold settings are required for reporting. Also the quantities for chemicals, if reported, are in broad ranges; it may not be possible to tell actual quantity. Therefore, a local resource (such as the fire department) is sometimes the best resource. Fire departments receive the same information as the LEPC but possess a broader knowledge of the community and smaller facilities.

Information on hazardous materials transported throughout the state via the highways can be obtained from the SERC or the state transportation department.

35 Some of the guidance presented here that is related to Regulatory Guide 1.78 is contained in NUREG/CR-6624 (Reference 3)

The same agencies may have information on the transport of hazardous materials via railways. The railways should also be contacted directly. Information on the transportation of chemicals via rivers, the Great Lakes and coastal marine traffic can be obtained from the U.S. Coast Guard.

Internet sources of data on hazardous materials available at the time this appendix was written include the following:

LEPC/SERC contacts:
www.rtk.net/lepc

Toxic release information:
www.epa.gov/tri

Hazardous substances profiles:
www.epa.gov/ceppo/ep_chda.htm#ehs

RMP data:
www.epa.gov/enviro

Right-to-Know data:
www.rtk.net or www.scorecard.org

Material Safety Data Sheets:
www.hazard.com

4.1.2 On-Site

A facility's EPCRA and RMP reporting information is a good first step to determine the types and quantities of hazardous materials on-site. This information should be compiled with a site-wide "walk through" using as a checklist the list of EPCRA and RMP hazardous chemicals. The checklist should be compared against a recent chemical inventory, which can usually be supplied by a facility department like purchasing, chemistry or stores. The walk through should also emphasize identifying permanent or temporary use of bulk storage containers or tanks such as propane as well as storage of asphyxiates like nitrogen and carbon dioxide.

4.1.3 Toxic Limits

The hazardous chemical toxicity limits that can be used for control room evaluations include those listed in Table C-1 of Regulatory Guide 1.78 or the IDLH exposure levels published by the National Institute for Occupational Safety and Health. Note that DG-1087 presents toxicity limits that may differ from those presented in Regulatory Guide 1.78.

The IDLH limits are based on 30-minute exposure levels defined as likely to cause death or immediate or delayed permanent adverse health effects. For the purposes of conducting control room habitability evaluations, the IDLH limits should be considered as toxicity limits for two-minute exposures. This provides an adequate margin of safety in that control room operators are expected to avail protective measures within two minutes of detection of hazardous chemicals, thus avoiding prolonged exposure at the IDLH concentration levels.

Asphyxiating chemicals should also be considered, if they are stored on-site in significant quantities such that an accidental release could result in the displacement of a significant fraction of the control room air. According to OSHA Regulations, an oxygen deficient atmosphere (for permit-required confined spaces) is one containing less than 19.5 percent oxygen by volume (29 CFR 1910.146).

4.2 Evaluating Potential Accidents

Whether a hazardous chemical source constitutes a hazard requiring a toxic gas control room evaluation is determined on the basis of the quantity of chemicals, the distance from the plant, the in-leakage characteristics of the control room and the applicable toxicity limits.

Section 4.2.1 presents screening criteria adopted from Regulatory Guide 1.78 for identifying release events that can be exempted from a detailed evaluation of control room habitability. For release events not meeting the screening criteria, Section 4.2.2 provides a basis for performing detailed evaluations of control room habitability.

4.2.1 Screening Criteria

Hazardous chemicals that meet the following criteria can be excluded from a toxic gas control room evaluation.

- *Distance Criterion for Stationary Sources*
  - Hazardous chemicals that are stored at distances greater than five miles from the plant can be excluded from a detailed toxic gas control room evaluation.
  - For those hazardous chemicals stored within a five-mile radius of the plant (except those hazardous chemicals stored in weights greater than 100 pounds either on-site or within 0.3 miles of the control room), Table C-2 of Regulatory Guide 1.78 gives the criterion in terms of the quantity of chemicals that would constitute a hazard for a given toxicity limit, stable meteorological conditions and a control room envelope with a given air exchange rate.[36]

- *Distance Criterion and Frequent Shipment Criterion for Mobile Sources*
  - Hazardous chemicals that are transported at distances greater than five miles from the plant can be excluded from a detailed toxic gas control room evaluation.
  - Frequent shipments are defined as 10 or more total shipments per year for truck traffic, 30 or more total shipments per year for rail traffic or 50 or more total shipments per year for barge traffic. Mobile sources need not be

36 Appendix A to Regulatory Guide 1.78 contains a procedure for adjusting the quantities given in Table C-2 to appropriately account for the toxicity limit of a specific chemical, meteorological conditions of a particular site and air exchange rate of a control room.

considered further if the total shipment frequency of all hazardous chemicals does not exceed the specified number by traffic type.

If the above screening criteria are not met, detailed evaluation as discussed in the following subsection should be performed to show that the control room is habitable in the event of an accidental hazardous chemical release.

4.2.2 Detailed Evaluations

An existing toxic gas evaluation should be revised if: (1) the assumed in-leakage value is found to be non-conservative; (2) a new significant source of hazardous chemical is identified in the vicinity of the plant; or (3) the quantity of chemicals is greater than previously assumed.

For each chemical considered, the value of importance is the maximum concentration that can be tolerated for two minutes without inducing physical incapacitation (i.e., severe coughing, eye burn or severe skin irritation) of an average human. The two-minute criterion is based on the time a control room operator is expected to take to don respirator and protective clothing. As stated in Section 4.1.3, the two-minute toxicity limit is based on the exposure limits.

If detailed calculations show that the two-minute toxicity limits will be exceeded in the control room for any time period for any given release scenario, compensating measures should be implemented.[37] As a minimum, a detection mechanism for each hazardous chemical release should be available. Such a system could include the installation of detectors or, if the buildup of the hazardous chemical in the control room is at a slow rate, human (i.e., smell) detection may be appropriate.[38] The detailed evaluation should demonstrate that if detection results in placing the control room in accident mode (i.e., automatic or manual closure of isolation dampers), the two-minute toxicity limits will not be exceeded. Otherwise, it would be expected that the control room operators will take protective measures (i.e., don protective equipment) within two minutes after the detection to avoid prolonged exposure at the two-minute toxicity limit levels.

There are several aspects that should be modeled when performing detailed evaluations of control room habitability due to potential accidental toxic gas releases: accident type, release characterization, atmospheric dispersion and control room air infiltration.

37 Compensating measures are not required for transportation-related accidents if it can be shown that the probability of occurrence of the initiating events leading to control room concentrations exceeding toxicity limits are less than 10^{-7} per year as discussed in Sections 2.2.1-2.2.2 of NUREG-0800.
38 The American Industrial Hygiene Association has established odor thresholds for a number of toxic chemicals. Some of these data are presented in NUREG/CR-6624.

- **Accident Type**. Two types of industrial accidents should be considered for each source of hazardous chemicals: maximum concentration accidents and maximum concentration-duration accidents.

  - For the *maximum concentration accident*, the quantity to be considered for each chemical is the instantaneous release of the total contents of one of the following:

    1) the largest storage container failing the screening criteria outlined in Section 4.2.1;

    2) the largest shipping container (or for multiple containers of equal size, the failure of only one container unless the failure of that container could lead to successive failures) failing the screening criteria outlined in Section 4.2.1; or

    3) the largest container stored on-site (normally the total release from this container unless the containers are interconnected in such a manner that a single failure could cause a release from several containers).

  - For the *maximum concentration-duration accident*, the continuous release of hazardous chemicals from the largest safety relief valve on a stationary, mobile or on-site source failing the screening criteria outlined in Section 4.2.1 should be considered.

- **Release Characterization.** The release characterization defines the physical state of the chemical as it leaves its containment and the manner in which it enters the atmosphere to form a vapor cloud. Since hazardous chemicals may be stored under pressure or under refrigeration, they can be emitted from a container as a liquid, a vapor or both, depending on the chemical's physical properties. For example, released liquids may form a vapor cloud through volatilization. A liquid can be volatized either completely or partially as it is released, forming a vapor cloud or a vapor and droplet mixture. Conversely, chemicals stored as a gas may partially or completely condense to form liquid droplets when released. Condensed vapor may fall to the ground to form a pool that, in turn, volatizes to the atmosphere.
- **Atmospheric Dispersion:** The resulting plume may be positively buoyant, neutrally buoyant or denser-than-air, based on the initial contaminant density compared to air. For dense gas releases, consideration can be given to modeling the release using a dense gas model; otherwise, standard passive dispersion modeling should be applied.
- **Control Room Air Infiltration:** The air flows for infiltration, makeup and recirculation should be considered for both normal and accident conditions. The volume of the control room and all other rooms that share the same ventilating air, during both normal conditions and accident conditions, should be considered.

Regulatory Guide 1.78 can be consulted for more specific details concerning performing evaluations of control room habitability for potential toxic gas releases.

Regulatory Guide 1.78 suggests using algorithms presented in its Appendix B for performing atmospheric dispersion modeling for instantaneous (puff) releases and algorithms presented in Regulatory Guides 1.3 *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors* and 1.4 *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors* for performing atmospheric dispersion modeling for continuous releases. Other options for performing atmospheric dispersion modeling analyses for hazardous gases whose densities are not significantly different than air include using Murphy and Campe (Reference 5.10) for releases near the control room (within 100m or so) and Regulatory Guide 1.145 *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants* for releases further from the control room.

NUREG-0570 *"Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release,"* is another accepted source of information for performing control room habitability evaluations. NUREG-0570 presents algorithms for calculating the fraction of a toxic release that flashes, along with algorithms for determining the evaporation rate of the remaining pooling liquid. Guidance for determining atmospheric dispersion and subsequent toxic gas buildup in the control room is also provided.

The NRC sponsored the development of a computer code system for evaluating control room habitability called HABIT (References 5.13 and 5.14). Two of the HABIT program modules, EXTRAN and CHEM, can be run in sequence to predict chemical concentration and exposures in the control room. The EXTRAN program computes atmospheric chemical concentrations associated with a release of a toxic chemical and the CHEM program use the results of EXTRAN to determine the associated chemical exposures in the control room.

In executing EXTRAN, the user should be aware of the following:

- EXTRAN does not calculate release rates and, as such, the user must calculate the release rate outside the model for the *maximum concentration-duration accident.*
- Regulatory Guide 1.78 suggests the atmospheric dilution factors to be used in the analysis should be that value which is exceeded only 5 percent of the time. Although EXTRAN uses a simple Gaussian dispersion model, the concentrations predicted by the model do not vary inversely with the wind speed because building wake correction is not a linear function of wind speed. In the case of evaporation, the highest emission rates are also related to high

wind speeds. In addition, the building wake corrections are not particularly sensitive to atmospheric stability. Consequently, a range of meteorological conditions should be executed for determining the 5 percent atmospheric dilution factors.

Several references describing methodologies for calculating release characterizations (including release rates) include EPA's "Workbook of Screening Techniques for Assessing Impacts of Toxic Air Pollutants" (Reference 6.15), "Risk Management Program Guidance for Offsite Consequence Analyses" (Reference 6.16) and "Guidance on the Application of Refined Dispersion Models to Hazardous/Toxic Air Pollutant Releases" (Reference 5.17). The latter reference also provides guidance on how to execute several generally available dense gas atmospheric dispersion models.

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16. USEPA, "Risk Management Program Guidance for Offsite Consequence Analysis," EPA-550-B-99-009, U.S. Environmental Protection Agency, December 1992.

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APPENDIX H

SYSTEM ASSESSMENT

1. PURPOSE

This appendix provides guidance on performance of walkdowns and inspections of the control room envelope and associated ventilation systems to identify potential vulnerabilities to in-leakage.

2. SCOPE

These guidelines assist personnel in the performance of assessment activities of the control room envelope and the associated ventilation systems with the intended purpose of:

(1) identifying potential vulnerabilities to in-leakage into the control room envelope,

(2) determining whether the system is configured and will align in a manner consistent with its licensing basis,

(3) identifying areas where maintenance activities should be directed,

(4) determining whether the control room envelop (CRE) and adjacent area ventilation systems are performing in a manner consistent with their licensing basis.

This appendix does not provide guidance for minimizing in-leakage vulnerabilities. Appendix J provides the guidance for minimizing vulnerabilities and sealing once the in-leakage source is identified.

3. ASSESSMENT METHODOLOGY

3.1 Identify the Boundary

This section ensures that the user has a good understanding of the boundaries and performance requirements for the control room envelope (CRE) and the ventilation system(s). The following process is recommended:

3.1.1 Obtain copies of the controlled as built drawings (e.g., flow, physical, general arrangement, etc.) that show the envelope and surrounding areas, the envelope ventilation system(s), and ventilation systems that traverse the envelope boundary.

3.1.2 Highlight the following on the drawings. This may require more than one set of drawings if the system response is different for different types of events:

- Boundaries of the envelope,
- Boundaries of the ventilation system(s) that serve the CRE,
- Portions of the ventilation system(s) that are physically located outside the boundary or perform a boundary isolation function (e.g., dampers). This should include system alignments for response to both radiological and toxic gas events, and
- Ventilation system(s) that traverse the CRE boundary. (Highlight and label on the drawings the routing of other ventilation systems that traverse the envelope.)

3.2 Identify Operating Configurations

Control room in-leakage must be measured with affected systems in their accident configuration. The next step is to determine whether the ventilation systems can be operated in the accident modes consistent with the licensing bases. Review available documentation (UFSAR, Operating Procedures, Emergency OPS, Abnormal Ops, etc.) to determine the various modes of operation. See Appendix I Section 5.2 for additional guidance with regards to operating modes for the systems.

3.2.1 Identify Operating Parameters

Establish the design performance parameters for the ventilation systems for the different challenges (radiological, toxic gas). These parameters include but are not limited to differential pressures, makeup and recirculation flow rates, duct static pressures and filter differential pressures. The purpose of this activity is to verify that the as-built system is consistent with controlled documents and to identify portions of the CRE that are at lower pressure than the surrounding areas. Identify ductwork of non-CRE HVAC systems that traverse the envelope and are at a higher pressure than the envelope. If this was done earlier as part of the design bases review for other sections of this document, simply refer to that work.

3.2.2 Consider the Challenges

During review of the pressures in the envelope and adjacent areas, consider all accident configurations of the control room ventilation system and of the ventilation systems in adjacent areas. A typical item that has been overlooked in the past and that should be factored into this review is the determination of automatic and/or manual response of the system to different challenges (examples: LOCA, FHA, MSLB, SGTR, and Toxic Gas). For example:

- A control room envelope could be pressurized during a radiological event and not pressurized during a toxic gas event.
- Operator actions taken per operating procedures during post-accident mitigation to realign ventilation systems can result in system alignments different than configurations due to automatic starting signals. Section 5, Comparing Existing Plant Configuration and Operations with Licensing Basis for CRH, provides more detailed guidelines for ensuring that operating procedures are consistent with design and analyses.
- The response of ventilation systems in adjacent areas can be different for a safety injection (SI) event vs. a control room high radiation event (non-SI event).

3.2.3 LOOP vs. a Non-LOOP Event

Ventilation system alignments serving the envelope and serving adjacent areas should consider the most limiting configurations. Consistent with the licensing basis for the facility, the user may consider a loss of off-site power (LOOP) coincident with the event. A LOOP is typically assumed to occur concurrent with an accident, but not with a toxic gas release. However, the user should recognize that assuming a LOOP coincident with the event may not provide the limiting condition for control room in-leakage. For example, ventilation systems in adjacent spaces may continue to operate during a non-LOOP situation and result in a less favorable differential pressure condition across the control room boundary. In other words, if the assumption of a LOOP results in the envelope being positive to all adjacent spaces, it may be more conservative to assume a non-LOOP event. This would need to be factored in with the overall accident response.

The CR operator analysis should be performed in a manner that maximizes the dose. The consideration of a LOOP may provide the highest in-leakage but not the highest calculated dose. Other factors need to be considered for the determination of dose.

3.2.4 Single Active Failure

Consideration of single active failures should be consistent with the licensing basis for the facility. Cases may exist where assuming all trains function as designed (i.e., no single failure occurs) could be more limiting from an in-leakage perspective. For example:

- For a neutral pressure control room, running both trains can result in an increased number of rooms within the control room envelope that have negative pressure relative to the adjacent areas.
- For a positive pressure control room, running both pressurization systems can result in increased unfiltered in-leakage if the fans are located outside the envelope.

3.2.5 Seasonal or Diurnal Changes

The alignment of ventilation systems and the corresponding pressures in the adjacent compartments (from those alignments) can also be sensitive to the time of year or the time of the day. That is, during different seasons or different times of the day, the ventilation systems serving these areas may be operated in different configurations depending on such things as outside air temperature.

3.3 Performing the Walkdown

There are several methods available to determine potential leak locations. Some of these are described below. These methods do not provide quantitative methods for determining in-leakage; they only aid the user in determining potential locations for in-leakage.

The walkdown should:

- Confirm that all components can be configured in their accident modes
- Verify that the normally indicated system parameters in the various operating configurations are consistent with the design and licensing parameters
- Verify the proper operation of ventilation systems adjacent to the control room boundary for the various challenges.

Section 3.4, below, provides a more detailed discussion of the types of items to consider during these inspection activities.

3.3.1 Visual Examination

During the walkdown, the inspectors need to be very deliberate in looking at details. Numerous small openings can yield relatively large leakage rates. The visual examination consists of a thorough walkdown of both the inside and the outside of the envelope boundary, where practical, to determine the physical condition and identify any unwanted openings. Specific areas to be visually inspected are identified in Section 3.4

Tools such as smoke pencils can be helpful to determine if leakage exists. Smoke pencils should be used deliberately to distinguish between a leak and random air currents (see guidance in ASTM E 1186).

Out-leakage may affect the ability of a positive pressure system to sufficiently pressurize the envelope. Out-leakage requires additional makeup air to maintain the positive pressure; even though this air is usually filtered, it still affects radiological and toxic gas assessments.

It should be noted that only easily accessible or large in-leakage sources are likely to be identified via walkdown.

3.4. Specific Considerations

Table H-1 provides a list of items to consider when evaluating potential vulnerabilities to control room in-leakage. The items in the table are applicable to several different potential system and envelope configurations, but not all of these may be applicable to any given plant. Table H-1 is not to be considered an all-inclusive list but only as guidance for the types of potential vulnerabilities. It may be helpful to list the vulnerabilities by type (e.g., doors, dampers, structural joints, etc.) and rank them in order of importance or suspected leakage.

The additional description below is to aid the user in the use of the Table H-1.

3.4.1 Control Room Ventilation System

For portions of ventilation systems located outside the control room envelope:

- CR ventilation systems that are located outside the control room envelope can experience in-leakage if portions of these systems (e.g., return ducting) are at a negative pressure relative to the area(s) they pass through.
- Some ventilation ducting (commercial, pocket lock, non-seal welded, non-bolted connections, etc.) can be a source of potential leakage

locations. Insulated ductwork can be difficult to inspect but can be a leakage source. If the ducting is a potential leakage source, the insulation may need to be removed to facilitate inspection.

- Air Handling Unit (AHU) housings can be a source of in-leakage if they are not welded or their integrity is compromised. For example, the underside of the housing can be a location of corrosion due to moisture accumulation.
- AHU electrical and instrumentation penetrations can be a source of unfiltered in-leakage.
- AHU and ventilation system doors, hatches, etc., can be a source of unfiltered in-leakage. Inspect such items as latches, sealing surfaces, seal compression, etc.
- Fan shafts can be a source of in-leakage if not sealed. This is due to the negative pressure at the fan shaft location.
- Loop seals and drains can be a source of in-leakage.

For portions of ventilation systems located inside the control room envelope:

- Portions of pressurization ductwork upstream of the filter and within the control room envelope can be a potential source of in-leakage. This portion of the system may operate at a higher pressure than the pressure in the envelope.
- Ducting that is isolated can be a source of unfiltered in-leakage if the isolation dampers are not leak tight. Typically this is a concern if the ductwork interfaces with the suction side of a fan (recirculation, AHU, etc.).

3.4.2 Other Ventilation System Ducting Within the Control Room Envelope

Ducting associated with other ventilation systems may be routed through the control room envelope. These can be a source of in-leakage if the system(s) operate at a higher pressure than the pressure within the envelope. Also note that control room pressure (or in some cases no pressure – example: isolation only for a toxic gas event) can influence the leakage from this ducting such that the lower the control room pressure the more the duct leaks. As an alternative to duct sealing or replacement, it is acceptable to change the operating mode of the subject ventilation system or secure it to ensure that it operates with a lower pressure than the envelope pressure. Isolating the ducting during post-accident mitigation does not exclude it from being a source of in-leakage because damper leakage in isolated ductwork may provide a potential source of in-leakage.

Ventilation ducting (commercial, pocket lock, non-seal welded, non-bolted connections, etc.) can be a potential leakage location. Seal welded ductwork should be visually inspected to ensure the integrity of the welds. Insulation may need to be removed from the ductwork to facilitate inspection to locate leaks.

3.4.3 Control Room Envelope Boundary Penetrations

- Penetrations such as cables and conduits, small pipes, etc., can be a potential source of in-leakage. To the extent practical, both the inside of the conduit and the conduit/wall penetration should be inspected for proper sealing.
- Other items such as concrete anchors through block walls, if not sealed, can be a leakage source at the interface.
- Ventilation equipment drains, system drains, floor drains, etc., commonly penetrate the envelope boundary. To prevent leakage through these lines, check valves or loop seals should be installed. If used, verify that the check valve design is appropriate for this application and the loop seals are maintained to keep them filled.

3.4.4 Doors in Control Room Envelope Boundary

Door seals can be a potential significant source of in-leakage. Experience has indicated that the door to door frame (sides and top of door) and the floor (bottom of door) can be significant leak locations. The inspection should ensure not only the integrity of the seals but verify that the door is properly compressing the seals.

3.4.5 Ventilation System Isolation Dampers

Control room ventilation system isolation dampers that close to ensure the integrity of the system and the envelope during an event can be potential sources of in-leakage. Redundant dampers should exist at each location to meet single active failure criteria.

Louvered dampers have historically proven to be unreliable isolation devices. This does not mean that these types of dampers are unacceptable, but louvered dampers are more susceptible to leakage than other designs. This does not imply that other types of isolation dampers cannot be a source of leakage.

Leakage can also occur through damper shafts or other associated sub-components that penetrate the ducting pressure boundary.

3.4.6 Other Non-HVAC Systems in the Envelope

Instrument air and/or service air systems can enter the envelope to provide air for damper controls, breathing air, etc. The compressors for these systems may be located outside the envelope and provide a means of unfiltered in-leakage if the components inside the envelope leak, or venting of air is part of the component operation.

Radiation monitors outside the envelope that draw samples from inside the control room envelope can be a source of in-leakage if the sample lines leak.

3.4.7 General Boundary Construction

Certain construction configurations or deficiencies are more susceptible to in-leakage. For example, porous (non-filled) block walls can leak, where poured intact concrete walls should not leak. Deficiencies such as cracks or inadequate sealing materials can be locations for in-leakage. Deficient expansion joints can be a source of leakage.

Areas that have been overlooked are those that are not readily visible; e.g., above dropped ceilings, below raised floors, against walls behind panels, etc. These should be inspected to the extent practical. In some cases, it may be easier to verify the boundary by looking at the other side.

3.4.8 System Flow Measurements

Airflow rates should be measured to ensure that the system flow rates are as expected for the various configurations. This document does not provide guidance on determining system flow rates. These measurements must be obtained from test results and compared with applicable limits to ensure that control room HVAC and interfacing systems are operating as designed. Ensure the tests were performed within appropriate time frame and represent current system parameters.

Significant discrepancies in air flow rates (i.e., the sum of the individual flow rates do not equal the whole) need to be evaluated. These types of conditions indicate the possibility for leakage and unwanted airflow. Differences may also be due to the uncertainty of the measurements.

The ventilation system should be properly balanced to ensure that ventilation flow rates are consistent with the design basis and to enhance pressurization (pressurized control room) or minimize differential pressures across the envelope boundaries (neutral control room).

4. Documentation

Document the control room boundary, the modes of operation, and the walkdown results including any in-leakage vulnerabilities (list vulnerabilities identified).

This information is to be used in performing testing per Appendix I.

Table H-1 Determination of Vulnerability Susceptibility

System / Component	Determining In-Leakage Vulnerability
Control Room Ventilation System Operation (Section 3.3.2)	Determine that ventilation systems are properly balanced Determine that ventilation system air flow rates and air sources are as expected
Control Room Ventilation System Integrity (Section 3.4.1)	Determine if control room ducting and/or HVAC equipment located outside the envelope is at a negative pressure with respect to adjacent areas. This is applicable to both operating and non-operating equipment. If this condition exists, consider the following vulnerabilities: • Ductwork including previous repairs with RTV sealant • Bellows, flanged and flexible joints • Equipment housings • System penetrations such as chiller lines, electrical and instrumentation • Accesses such as doors or hatches • Fan shaft (AHU, Recirculation fan, etc). Determine if portions of the pressurization ducting inside the envelope between the envelope boundary and the filter are operated at a higher pressure than the envelope pressure (for portions of the ductwork located inside the envelope). Determine if AHU fans have the potential to draw air from isolated ducting lines (i.e., damper leakage) that penetrate the envelope boundary.

Table H-1 DETERMINATION OF VULNERABILITY SUSCEPTIBILITY

System / Component	Determining In-Leakage Vulnerability
Other Ventilation Systems (Section 3.4.2)	Determine if other system ducting is routed through the envelope when the control room is isolated. If so: • Determine the post-accident pressure in the ducting relative to the pressure in the envelope (consider the effects of this ducting both as a means of in-leakage and out-leakage). • If the ducting is isolated, consider the potential for damper leakage. • Determine the integrity of this ducting. Consider the items identified above under CR HVAC integrity.
Penetrations in the Envelope Boundary (Section 3.4.3)	Determine that wall, floor and ceiling penetrations (i.e., conduits, electrical cable trays, etc.) are properly sealed. Check for voids inside cable bundles that may be covered with cable coating or voids under the cable in the tray. Check for non-leak-tight flexible conduit or armored cables passing through penetration seals. Check seals inside the conduit and between the conduit and the wall. Check conduit connectors, couplings and terminations. Check caps on spare embedded sleeves. Determine that ventilation ducting penetrations and dampers are properly sealed. Check for space around fire damper sleeves. Check for concrete anchors or other bolts through block walls that are not sealed. Determine that drains (floor or equipment) have loop seals or check that valves and abandoned drains are sealed. If used, verify that the check valve design is appropriate for this application. Determine if there are other types of penetrations that can provide potential leakage pathways.
Envelope Doors (Section 3.4.4)	Determine that there are no defects in the doors. Determine that door seals (including sweeps) are not cracked, are not missing and have proper fit. Determine that doors are properly compressed or fitting against the door seals.

Table H-1 DETERMINATION OF VULNERABILITY SUSCEPTIBILITY

System / Component	Determining In-Leakage Vulnerability
	Determine that door latches are functioning properly to maintain the door securely closed. Determine that doorframes are properly sealed.
Isolation Dampers (Section 3.4.5)	Determine that control room isolation damper seals are not cracked, are not missing seals and have proper fitting seals. Determine that control room isolation damper linkages are functioning properly to assure compression of the seals against the damper blade(s). Determine that damper shaft penetrations are properly sealed.
Other Non-HVAC Systems in the Envelope (Section 3.4.6)	Determine if there are instruments or service air lines that enter the envelope boundary and could provide potential unfiltered air sources due to leakage or operational venting of air operated components. Consider other equipment operations providing a mechanism for air in-leakage such as radiation monitors that are located outside the envelope and draw a sample from within the envelope.
General Boundary Construction (Section 3.4.7)	Determine that the general envelope boundary is in good condition, including: • Block walls – unsealed or unpainted, cracked or missing mortar • Metal deck – joints and ceiling interfaces with walls • Plaster or drywall – unsealed over armor plate • Steel/concrete interfaces – structural steel, doorframes • Concrete – cold joints, expansion joints, seismic gaps • Hidden or abandoned chases or spaces or joints hidden under carpet. Fireproofing - penetrating envelope or covering joints or penetrations

APPENDIX I

TESTING PROGRAM

1. Purpose

This appendix provides guidance on preparing for and performing control room envelope in-leakage tests to demonstrate conformance to the plant licensing and design bases.

2. Scope

This appendix focuses on conducting a test that will quantify in-leakage into the control room envelope. The guidance includes the attributes of an acceptable test program, acceptable testing options, preparation for testing, performance of testing, and test frequency. This appendix is intended to aid plant personnel in the development of a plant specific procedure for testing.

3. Regulatory Basis

Criterion III of 10 CFR 50 Appendix B requires that design control measures provide for verifying or checking the adequacy of design. One of the methods identified to accomplish this for control room integrity is the performance of a suitable testing program.

4. Test Attributes

The attributes of an acceptable test program are:

- The test must be comprehensive.
- Integrated system testing must be conducted with systems and components in their accident configuration lineups. Component testing must be conducted in a manner that bounds accident conditions.
- Testing methods should be performed using a consensus standard.

4.1 Comprehensive

Any acceptable test method should be comprehensive. A test is considered comprehensive if it quantifies all of the in-leakage associated with a control room envelope. A comprehensive test program will determine the total control room envelope in-leakage for each challenge (e.g., toxic gas, radiological) that may be encountered. Some plant designs may be such that the control room envelope ventilation system(s) and associated components function in the same manner regardless of the challenges. In those cases, the results of one test will identify the leakage associated with both challenges.

4.2 Configured and Operating

Test conditions should be consistent with the limiting conditions in the licensing basis.

Tests that are performed to determine control room envelope (CRE) in-leakage should be performed with the envelope, its associated ventilation systems, and ventilation systems in adjacent areas all aligned and functioning the way they would if a radiological or toxic gas event were to occur. Alternatively, individual leakage sites may be tested with the ventilation systems in a non-accident alignment providing the test conditions for the components are representative of the accident condition. For example, damper leakage may be tested in a static condition as long as the ambient temperature and pressure differential are at the level expected with the accident ventilation systems operating.

4.3 Industry Standard

Tests that demonstrate control room envelope integrity should be performed using a recognized industry standard. The industry standard should be relevant to the determination of in-leakage.

5. Testing

This section provides guidance on developing a test program and choosing an appropriate test method.

5.1 Prerequisites to Testing

- An assessment of the control room boundary should be performed in accordance with Appendix H of this document. This includes the walkdown portion of the appendix plus any sealing/refurbishment/repairs needed as identified in the assessment.

- Determine acceptance criteria.

- Plants should have contingency plans in place to address results that may challenge the operability of the control room ventilation system. Development of contingency plans should include calculations on maximum allowable radiation in-leakage, maximum allowable radiation in-leakage for operability determinations, and maximum allowable toxic gas in-leakage. (Appendix C provides analytical guidance on the calculations, Appendix F has guidance on compensatory measures.)

- HVAC systems (including adjacent spaces HVAC systems) should be properly aligned and balanced to meet air flows and pressures consistent with the licensing basis. This information is verified in Appendix H.
- The impact of other plant activities on the test, and of the test on other plant activities, should be assessed. The ingress and egress of the control room boundary may need to be limited during the test.
- Plants that use outside air for pressurizing their control rooms will still need to continue to verify that the amount of pressurizing air is within acceptable limits.

5.2 Determine System Mode of Operation for Testing

Two common modes of operation are pressurization (isolation with pressurization) and isolation (isolation without pressurization). The pressurization mode is generally for protection from radiological events and the isolation mode is generally for protection from toxic gas events. However, this varies among plants and the possible system alignments that need to be tested should be carefully determined by each licensee.

Testing should be performed with a sufficient number of different system modes of operation to verify the adequacy of the system for all design basis events. For example, if the plant has a licensing basis toxic gas event that results in a required isolation of the control room, the system should be tested in the isolated mode and in-leakage determined (this includes not only the HVAC serving the control room but also adjacent spaces HVAC).

If the plant can show that one test configuration encompasses all operational configurations (i.e., the mode being tested will yield the highest in-leakage value and this value can support all applicable analysis) then multiple tests are not required[39]. The system modes for testing should be documented along with the basis for the system mode tested.

5.3 Determine Method of Baseline Testing

In-leakage baseline values must be determined for control room envelopes (CREs) for radiation dose considerations and toxic gas concerns if applicable. This section provides guidance on two methods of baseline testing and allows for alternative tests

39 For the case of a plant designed for positive pressure to radiation but neutral for toxic gas, leakage through the envelope boundaries in the neutral configuration can be either in or out, depending on the direction of the differential pressure. Therefore, performing two separate tests should be considered for the toxic gas and the radiological control room response.

that may be applicable in certain situations. The first method determines total leakage into the control room envelope by an integrated tracer gas test. The second method determines control room in-leakage by testing individual components and summing the results to obtain total leakage.

The method of testing selected depends on the vulnerability of the CRE to in-leakage. For neutral pressure control rooms or for positive pressure CREs with a large number of vulnerabilities or where testing a specific component vulnerability is not feasible, an integrated tracer gas test is recommended. For plants with positive pressure control rooms, small assumed in-leakage values, minimal vulnerabilities, and where methods to test the vulnerable components are feasible, component testing may be an acceptable method. The plant may perform an economic evaluation of the different acceptable test methods to determine the optimum choice. The type of testing that is to be performed (tracer gas, component, alternate) must be documented along with the basis for the test chosen. Sections 5.3.1, 5.3.2 and 5.3.3 provide additional information that will assist in determining the type of testing to be performed.

5.3.1 Tracer Gas Testing

For control room envelopes that are not positive to all adjacent areas or have numerous vulnerabilities to in-leakage, tracer gas testing is likely the most effective test method. A number of plants in the nuclear industry have used this test method for measuring control room envelope in-leakage.

This test method determines total in-leakage of the CRE by one of three techniques. They are (1) concentration decay, (2) constant injection and (3) constant concentration. These techniques are described in ASTM E741, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution." Depending upon the technique, they involve the measurement of makeup flow to the control room envelope, the concentration of the tracer gas in the control room envelope, and the injection rate of the tracer gas. The concentration decay method has generally proven the most effective method for systems with no makeup while the constant injection technique has generally proven the most effective method for systems with high makeup flow rates for pressurization. These measurements result in the determination of the volume rate across the boundary. The in-leakage can be inferred from these measurements.

The tracer gas test may not identify the in-leakage contribution of individual components or the specific location of the problem areas. That is dependent upon the tester and the limitations and complexity associated with the facility to be tested. However, tracer gas testing has been used to measure the component leakage associated with dampers, shaft seals, etc.

ASTM E741 provides a description of the limitations associated with the tracer gas test and identifies the knowledge and expertise requirements of individuals using the test method. These aspects should be understood prior to the performance of any test. Consider the following items when performing a tracer gas test:

- This test is dependent upon ensuring uniform tracer gas concentration throughout entire control room volume and upon appropriate sampling techniques.
- Proper selection of the best measuring points for tracer gas test and injection points for tracer gas prior to test initiation is important to the success of this test method.
- Determination of the net volume of the control room envelope may also be important. This volume enters into the calculations of in-leakage. The more accurate the value, the more accurate the results of the tracer gas test.
- Effects of the environment on the test results should be considered. Performing the test to minimize environmental influence is recommended. The test instruction should contain guidance on environmental effects. For example, the test should not be performed if there is a strong consistent wind (>15 mph) and the control room envelope is significantly exposed to the outside environment. The lower the wind speed, the more accurate the test results.
- Because of test complexity, plants typically require outside expertise to perform this test.

All system testing within the scope of this appendix requires that systems be tested in their accident configuration lineup or in a configuration that will result in a conservative in-leakage measurement.

5.3.2 Component Testing

For positive pressure CRE designs with a small number of vulnerabilities to in-leakage, component testing may be an effective method for determining in-leakage. Control room designs with the following features support this method of testing:

- CREs that are maintained at positive pressure with respect to **all** adjacent spaces.
- Majority of control room HVAC equipment and ducting is located within the control room envelope.
- Minimal non-control room ventilation ducting or air system piping penetrate the control room envelope.
- Ventilation ducting located outside the CRE is of the seam welded design and is in good material condition.

This method is dependent upon a thorough assessment of the control room envelope boundary and ventilation systems to ensure all potential vulnerabilities are properly evaluated. Missed vulnerability results in indeterminate test results due to an undefined uncertainty. The use of independent peer industry personnel on the assessment team is a good practice. Thorough documentation of the assessment results is required for providing assurance that in-leakage vulnerabilities are not overlooked.

This test method relies on pressure or vacuum decay testing for measuring leakage. It is based on the fact that air moves from a region of high pressure to a region of low pressure. Table I-1 provides information on some industry testing standards that can be used for a component test.

Two steps are performed to quantify total CRE in-leakage.

- First, the CRE differential pressures must be measured in enough areas to ensure that the envelope is maintained positive with respect to adjacent non-CRE spaces. This provides assurance that any leakage through boundary walls floors, ceilings/roofs will be out-leakage.
- The second step is to test all components that were identified as vulnerable to in-leakage by the Appendix H assessment. These are component boundaries where differential pressure conditions can not be verified. The sum of the individual component in-leakage values will become the total unfiltered in-leakage rate.

Component testing results in identification of a total CRE in-leakage rate in CFM. A comprehensive assessment must accompany this testing to assure that all potential in-leakage pathways are tested. Each component test should be performed per an acceptable consensus standard such as those listed in Table I-1.

Component testing is to be performed in a manner that reflects accident configuration leakage. The test differential pressure across the component must reflect the differential pressure that would be observed in the component during accident conditions. The effect of HVAC systems in adjacent areas that may not operate in accident conditions must be accounted for when establishing component test conditions.

Component testing is probably within the capability of the plant staff.

This test method is not considered applicable to control room designs that are not pressurized in the emergency mode or where a large vulnerability to in-leakage may exist. Therefore it is important that a thorough evaluation per Appendix H is performed and that accurate differential pressure measurements are made.

5.3.3 Alternate Test Methods

Licensees may propose alternate test methods. Alternate test methods must meet the following criteria:

- Test all potential leak paths and produce an overall in-leakage value in CFM for the entire CRE.
- Performed in accordance with a consensus test standard such as those listed in Table I-1.
- Conducted in a manner that reflects accident configuration leakage.

Licensees that propose to measure in-leakage using an alternate test method should include a detailed description and justification of the proposed method to allow a knowledgeable reviewer to ascertain the acceptability of the test.

See the attached Table I-1 for methods that may be considered for development as an alternative test method. Note that a combination of methods may be necessary to produce an overall in-leakage value in CFM for the entire envelope.

5.4 Performance of Baseline Testing

Based on the determination made in Section 5.3 either Section 5.4.1 (tracer gas) or 5.4.2 (component) may be used. If an alternate test method is chosen then the utility should establish the guidance related to the alternate test.

5.4.1 Integrated Tracer Gas Testing

The industry standard currently being used for a tracer gas test to determine in-leakage is ASTM E741, *Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution.* It is beyond the scope of NEI 99-03 to provide a detailed procedure applying ASTM E741; however general guidance is presented in preparing and conducting the test.

After Sections 5.1, 5.2 and 5.3.1 above are completed a sequence of three steps is recommended to prepare for and conduct the integrated tracer gas test. These steps are:

- Determine if the test is to be performed in-house or using a contractor.
- Perform a walkdown of the CRE (including sealing of potential leak paths identified in Appendix H.
- Perform the test.

These steps are described in greater detail in Section 5.4.1.

5.4.1.1. Preliminary Actions

The following steps should be performed prior to performing a tracer gas test.

a. Determine the test configuration for the CRE, CR HVAC and adjacent space HVAC (i.e., damper positions, equipment lineups, control of personnel entry/exit, pressurization mode, etc.). If using a contractor this should be done prior to bringing them on-site.
b. Obtain Material Safety Data Sheets for the tracer gas for incorporation/approval by the site's material control program.
c. Determine the net volume of the CRE, if needed. This volume enters into the calculations of in-leakage. The more accurate the value, the more accurate the results of the tracer gas test.
d. Determine if the test is to be performed in house or by a contractor.
e. If a contractor is to perform the test then:
    - Ensure the contractor is familiar with this type of testing.
    - Determine if the contractor has a 10 CFR 50 Appendix B QA program. This will play a major role in deciding whose QA program will apply and whether the vendor can provide calibrated measuring and test equipment.
    - Familiarize contractor personnel with the plant configuration, the purpose of test and the control room HVAC mode to be tested prior to arrival on-site.
    - Review the CRE Boundary and CREVS configuration and operation (on-site) in detail with the tracer gas testing contractor identifying:
        1. test configuration(s)
        2. measured data required for habitability analysis
        3. CRE boundary and boundary condition walk-down
        4. CREVS configuration walkdown
    - Walkdown the CRE with contractor to select best measuring points and injection points for tracer gas prior to test initiation. This should be conducted with a set of as-built drawings.
    - Select the method of measurement that is appropriate for the CRE to be tested (examples: concentration decay, constant injection and constant concentration).
    - Verify that contractor test procedures are compatible with plant procedures (includes but not limited to):
        - Test equipment calibrations
        - Test personnel qualifications
        - Tracer gas test compatibility with plant chemical tracking program.

5.4.1.2. Performing the Tracer Gas Test

To the greatest extent possible the test conditions should correspond to the analysis of the system. The minimum time recommended to perform the test is provided in ASTM E741 as a function of the method of measurement.

Key factors affecting accurate testing are:

- Uniform mixing within a zone,
- Representative sampling (multiple samplers),
- Determination of CRE net volume and
- Measurement of pressurizing flow rate, if applicable.

Additional considerations for performing an effective test are:

- Follow all appropriate Technical Specification limiting condition for operation and plant operating procedures

Consider the effects of the environment on the test results consistent with the plant design basis assumptions. The test instruction should contain this guidance on environmental effects. For an example: the test should not be performed if there is a strong consistent wind (>15 mph) and the CRE is exposed significantly to the outside environment. The lower the wind speed, the more accurate the test results.

- Prepare plant specific test procedure (s) in accordance with plant requirements. The test procedure should allow for using the contractor's actual tracer gas test methodology (if a contractor was selected).
- Brief plant operations personnel.
- Consider including a requirement to limit door openings/closings during true test.
- Perform testing in accordance with plant procedures.
- Retest, if necessary.

5.4.2 Component Testing

This test is dependent upon the premise that the CRE is at a positive pressure to all adjacent areas; however, testing must validate this premise. In this respect, the differential pressure measurements are critical. These differential pressure measurements are used to demonstrate that there is only out leakage across *the boundary walls, floors and roofs/ceilings. This includes the doors and all* penetrations in the boundary. Any component that cannot be verified to have a positive differential pressure across the boundary must be tested for in-leakage. See Table I-2 for more discussion of the components and testing of those components.

5.4.2.1 Differential Pressured Measurements

Note: This step shows that the CRE is at a positive pressure and therefore it can be concluded that in-leakage will not occur across the CRE walls, floors and ceiling/roof.

Air flows from areas of high pressure to areas of low pressure. Thus, leakage through the envelope boundary occurs from the area(s) of high pressure to the area(s) of lower pressure. Therefore, it is crucial to determine the pressure(s) within the envelope relative to the adjacent areas outside the envelope boundary when identifying the potential sources of in-leakage. This is valid regardless of the ventilation system design (pressurized control room or neutral pressure control room).

For a positive pressure envelope design, leakage occurs outward from rooms within the envelope to adjacent areas provided the space in the envelope is at a higher pressure relative to the adjacent space. Even with a positive pressure CRE design, in-leakage can occur at walls, ceilings, floors, ventilation ducting, dampers, drain lines, from other systems that traverse the envelope, etc., unless those areas are shown to be at a positive pressure relative to adjacent spaces. Note that excessive out-leakage from the envelope should be minimized as this places an increased demand on the pressurization system and increases the filtered in-leakage value in the dose assessments.

To determine if there are any adjacent areas that could be at a higher pressure than the rooms within the CRE, a control room positive pressure test must be performed. This test measures the control room pressure relative to spaces adjacent to the CRE. The plant must identify acceptance criteria for an acceptable positive pressure. For adjacent spaces that are essentially outside atmosphere a positive 1/8 (0.125) inch water gage pressure differential is recommended to allow for atmospheric variation. For adjacent areas inside a building where conditions are more stable, a positive pressure of 0.05 inches water gage is sufficiently high enough to allow accurate measurements. Precision digital barometers can be used. Barometers of accuracy of +/-0.03 inches of water are readily available. Precision digital manometers capable of sensing pressure changes of 0.0001 psi are also available. The use of two precision instruments is recommended. The adjacent measurements should be timed and corrections made for elevation differences and other environmental influences between different spaces. Items to consider when measuring the differential pressure include:

- Use a drawing to identify all the control room areas and adjacent spaces to be measured.
- The system mode of operation when the pressure measurements are taken must be consistent with the modes of operation defined in Section 5.2 of this appendix.
- The preferable method is to measure with a differential pressure (d/p) gage for accuracy considerations. If a d/p gage is not available, measuring the pressures with a pressure gage is acceptable. If smoke pencils are used to show a positive pressure then it should be noted on the test report.
- Measure the pressures in <u>all</u> adjacent areas to the envelope. The control room positive pressure test must be done in sufficient areas to assure that a comparison is made with all adjacent areas.
- Measure the pressure in <u>all</u> rooms within the envelope. Take enough measurements within a given room to ensure that pressure variations in the room do not result in any negative pressures relative to adjacent non-CRE areas. For example, complicated room configurations with restrictions to air flow (panels, half walls, etc.) can result in pressure variations within the room. Elevation and temperature differences can also affect pressure differential and should be accounted for. All areas adjacent to the boundary must be represented by a pressure measurement.
- Care should be taken to measure pressures in hard to get areas such as above dropped ceilings or below raised floors to ensure that these areas are not at a negative differential pressure relative to adjacent non CRE areas.
- Record and compare the pressures of the adjacent spaces to the areas inside the control room boundary to show the control room is at a positive pressure to all adjacent spaces. The control room must be at a higher pressure than the adjacent spaces.
- Monitor outside pressure while taking differential readings across the CRE boundary. Many instruments are very sensitive and changes such as the passing of a weather front can inject significant changes in data readings.

If it is discovered that adjacent area(s) are at a higher pressure than the pressure inside the CRE, actions may be taken to reduce the pressure in the adjacent area. Ventilation system operating configurations should be considered as well as securing fans (if feasible) and providing pressure relief paths. This is addressed in more detail in Appendix J.

If the system is rebalanced or in any way changed such that the differential pressure measurements are affected, then sufficient additional measurements must be taken to assure that the CRE walls, floors, ceiling/roofs are still positive to all adjacent spaces.

If it cannot be shown that the CRE is positive relative to adjacent areas, then a component test cannot be performed and another test method must be used.

5.4.2.2 Determine Scope

Any component that cannot be verified to have a positive differential pressure across the boundary must be tested for in-leakage. Use the differential pressure measurements from Section 5.4.2.1 and Table I-2 to make this determination. Each vulnerability (i.e., component) identified in Appendix H must be addressed (some items such as doors may not need a test if the CRE positive pressure test confirms that any leakage would be out-leakage from the envelope). Record the components that are to be tested. Examples of components that could be tested individually are air handling units, ductwork and isolation dampers.

5.4.2.3 Select Test Method for the Component

Available methods for testing the leak tightness of components[40] are provided in Table I-2.

Document the type of component test that will be used for each component.

5.4.2.4 Perform the Applicable Test

- Perform each test as identified in 5.4.2.3.
- Record the leakage measurements made[41].
- Sum all the leakage measurements. This is the total unfiltered in-leakage.

40 Dampers that close when ventilation systems realign to the emergency mode such that the pressure inside the damper is negative with respect to the outside air may become a potential source of additional in-leakage into the control room envelope that can be filtered or unfiltered depending upon the damper location in the system. ANSI N510-1989 provides methods to test this leakage using a totalizing gas flow meter or possibly a calibrated rotating vane anemometer. Industry standard ASTM E 2029-99, "Standard Test Method for Volumetric and Mass Flow Rate Measurement in a Duct Using Tracer Gas Dilution," discusses the use of tracer gas on a component level by a constant injection at the damper air intake with measurements downstream of the closed damper. The constant injection method is considered advantageous in that control test volumes are not required that may require fabrication within the installed ductwork. Measurement uncertainties can be determined using ANSI Standard PTC 19.1, "Measurement Uncertainty."

41 For control room envelopes that can tolerate large amounts of unfiltered in-leakage, flow measurements are acceptable provided the measurements consider instrument error.

5.5 Test Results

- Document the components to be tested as identified in Section 5.4.2.2; if a component test is being performed.
- Document all test results including leakage measurements.
- Provide one value for in-leakage for each lineup tested. The test results must account for the uncertainties associated with performance of the test including the accuracy of the test equipment used.
- If measured values are higher than acceptance limit, compensatory measures may need to be taken to maintain the control room ventilation system operable until permanent resolution is achieved (See Section 8.0 and Appendix F for guidance). In-leakage values that result in doses greater than that currently reported in the UFSAR will require evaluation per the plant's corrective action program.

5.6 Documentation

Table I-3 identifies recommended documentation.

TABLE I-1
TESTING OPTIONS

Type of Test	Standard	Advantages	Disadvantages	Performed with systems in their accident configuration	Optimum accuracy	Quantitative
Tracer Gas (SF6) (Note 3)	ASTM E741	1. History of use within the industry 2. Test method acceptable to NRC	1. Wind effects 2. Disrupts plant 3. High cost 4. Leak location not identified 5. Tests from inside out	Yes	± 10% (Note 1)	Yes
Pressure Test (Blower Door)	ASTM E779	1. Performed under positive or negative press 2. Requires CR HVAC shutdown	1. Req. CR HVAC shutdown 2. Tests from inside out 3. No leak location identified 4. Impact on operations 5. Wind effects 6. Seal supply and return duct	No. CR HVAC is shutdown	± 5% (Note 2)	Yes
Leak Detection	ASTM E1186	1. Identifies location 2. Inexpensive 3. No effect on Operations	1. Cannot quantify leakage	Yes	N/A.	No.
Component Test	ASTM E779 ASTM E1186 ASTM E741 ASME N510 ASME AG-1 10CFR50, App J, Type C LLRT method (Note 4)	1. Potentially lower cost 2. Low impact on operations 3. Identifies leak location	1. Requires isolation of individual components	Section by section	Test dependent	Yes

Notes:

1. Tracer gas testing is comprehensive for neutral pressure control rooms but requires flow measurements for positive pressure control rooms, which increases the overall uncertainty of the test result. If the actual unfiltered in-leakage is small (< 100 CFM) and the pressurizing air flow is relatively large (>1000 CFM), the uncertainty in the air flow measurement causes the accuracy of the tracer gas test to become very poor (30% - 60%). Using the parenthetical numbers as an example, an uncertainty of 10 percent in the airflow measurement yields an error band of at least +/- 100 CFM. When this error is compared to the measured in-leakage, the overall test uncertainty approaches 100 percent measured.

2. Accuracy depends on how the flow measurement is made. Blower door testing provides an indication of structural leak tightness rather than a good measure of in-leakage. This test method does not measure air leakage rates under normal conditions of weather and building operation.

3. Testing developed by Haven Brook National Labs using multiple tracer gases has the potential for conforming to an acceptable test; but has not been researched for NEI 99-03. This method has the ability to discriminate and quantify leakage through different barriers (web site http:www.bnl.gov/ecd.htm). WEB SITE current as of 9/3/00.

4. The volume between closed isolation dampers installed in tandem can be pressurized and the volumetric flow required to maintain the test pressure measured as the leakage. One of the two dampers will be tested in the direction opposite the normal differential pressure condition. The results should be conservative since damper leakage in this direction should be greater than if it is tested in the normal differential pressure direction.

TABLE I-2
SELECTION OF COMPONENTS FOR COMPONENT TEST

Vulnerability Area	Discussion	Component Test Required/Not Required	Acceptable Component Test
CRE ceiling/roof	The positive pressure measurements of the CRE would show that this vulnerability would not exhibit in-leakage as the leakage would be out of the CRE.	Not required as positive pressure precludes in-leakage	NA
CRE walls	The positive pressure measurements of the CRE would show that this vulnerability would not exhibit in-leakage as the leakage would be out of the CRE.	Not required as positive pressure precludes in-leakage	NA
CRE floor	The positive pressure measurements of the CRE would show that this vulnerability would not exhibit in-leakage as the leakage would be out of the CRE.	Not required as positive pressure precludes in-leakage	NA
CRE penetration in roof/ceilings; walls; floor	This is the external portions of the penetrations. The positive pressure measurements of the CRE would show that the perimeter of these penetrations would not exhibit in-leakage as the leakage would be out of the CRE. This also includes other types of penetrations that can provide potential leakage pathways; for example, concrete anchors through block walls, which are not sealed.	Not required as positive pressure precludes in-leakage	NA
CRE doors	The positive pressure measurements of the CRE would show that this vulnerability would not exhibit in-leakage as the leakage would be out of the CRE.	Not required as positive pressure precludes in-leakage	NA
Electrical conduits	Determine that wall, floor and ceiling penetrations (i.e., conduits, electrical cable trays, etc.) are properly sealed internally. If the internals are not sealed then smoke pencils may be used to verify no leakage through the open conduit, etc. However, if there is flow	Not required provided that the conduits, etc. are properly sealed	NA, otherwise use smoke pencils. See discussion.

Vulnerability Area	Discussion	Component Test Required/Not Required	Acceptable Component Test
	indicated passing through the open conduits then an integrated tracer gas test may be required.	internally	
Ducting, housings located outside the CRE	Determine if control room ducting and/or HVAC equipment located outside the CRE is at a negative pressure with respect to adjacent areas. This is applicable to both operating and non-operating equipment, and to both HVAC ducting and filter system ducting. Any ducting and/or housings under a negative pressure are a potential source for in-leakage. Access doors, hatches, instrument lines, drain lines (should have loop seals to prevent leakage), damper and fan shafts,	Required	See Acceptable Method this table
Isolation dampers located outside the CRE and the ducting between the CRE wall/floor/ceiling and the damper	Determine if AHU fans have the potential to draw air from isolated ducting lines (i.e., damper leakage) that penetrate the envelope boundary. Dampers may leak at the damper seals plus the ducting may leak.	Required	See Acceptable Method this table
Ducting, housings located within the CRE	Determine if AHU fans have the potential to draw air from isolated ducting lines that penetrate the envelope boundary.	Required for ducting that is susceptible to in-leakage	See Acceptable Method this table
Isolation dampers within the CRE and the ducting between the CRE wall/floor/ceiling and the damper	Determine if AHU fans have the potential to draw air from isolated ducting lines (i.e., damper leakage) that penetrate the envelope boundary. Dampers may leak at the damper seals plus the ducting may leak.	Required	See Acceptable Method this table
Ducting passing through the CRE that is not isolated and is not part of the CR HVAC.	Determine if internal pressure of ducting is greater than the CRE. Ducting may leak into the boundary and be a source of in-leakage.	Required	See Acceptable Method this table

Other systems	Radiation monitors and pneumatic air airlines may be a source of in-leakage. These systems should be reviewed for leakage. Constant bleed air regulators can be a source of unfiltered in-leakage along with operational venting of air operated components.	Not required if it can be shown that the lines do not leak. For pneumatic air bleed of the maximum amount of design bleed of a component (continuous or as cycled) shall be used. No test required for this item.	NA

Component	Acceptable Method*
Dampers	1. Direct Measurement Method of ANSI N510 Standard 2. Tracer Gas Technique using ASTM E 2029 Standard 3. ANSI ANS-56.8, "Containment System Leakage Testing Requirements"
Ducting and housings	1. Direct Measurement Method of ANSI N510 Standard 2. ASME AG-1-1997, "Code of Nuclear Air & Gas Treatment," Section TA, Mandatory Appendix TA-III, "Duct and Housing Leak Test Procedural Guidelines"

*Other methods may be acceptable if they are associated with a standard. The methods presented above are already accepted by the industry and NRC for measuring leakage in ducts, housings and dampers.

Table I-3
Appendix I
Critical Documentation Requirements

Section	Type of Test Selected*		
	Tracer Gas	Component	Alternate
5.1 Acceptance Criteria Documented	Yes	Yes	Yes
5.2 System Mode (s) for Testing	Yes	Yes	Yes
5.3 Selection of Baseline Test - Basis	Yes	Yes	Yes
5.3.2 Components selected to be tested	NA	Yes	NA
5.4.2.4 In-leakage values measured for each mode selected in 5.2	Yes	Yes	Yes

NA - Not Applicable
* - One type of test should be selected for toxic gas, and one type of test should be selected for radiation. If the plant response to both events is the same, then only one test that covers both toxic gas and radiation need be performed.

APPENDIX J

CONTROL ROOM ENVELOPE SEALING PROGRAM

1. Purpose/Scope

The purpose of a control room envelope (CRE) sealing program is to monitor and maintain the pressure boundary penetrations such that the CRE habitability design and licensing bases are met and maintained.

2. CRE Barrier Control

Control of the CRE pressure boundary should be maintained at all times (see Appendix K). In the event that planned maintenance work, testing or plant conditions will affect the CRE boundary, administrative control of the boundary should be procedurally maintained.

3. Sealing Program Development

A CRE assessment, as outlined in Appendix H, should consider the vulnerability of the envelope to leakage. The assessment should include a review of applicable building and system drawings and walkdowns. This information can then be used to identify all penetrations, prioritize them according to safety significance and develop a cost-effective sealing program. Such a program should include required inspection frequency, type of acceptable materials, and repair and test procedures. The method and frequency of inspection/repair/modification will depend on the type and safety significance of the seal.

The following is a list of typical penetrations and/or items that may have seals that would allow in-leakage.

Abandoned pipe chases
AHU drains
AHU housing
Cable trays
Card readers
Conduits
Conduit penetrations
CR pressure boundary ducting outside CRE
CRE walls/ceilings/floors
Doors
Duct access panels
Duct expansion joints
Duct penetrations

Ducting traversing CRE and at higher pressure
Expansion joints or seismic gaps
Fan housing/shaft
Fire dampers
Filter housing/drains
Flanged joints
Gaps at building wall/floor/ceiling intersections
Instrument air lines supplying CRE pneumatic components
Isolation dampers / shafts and gaps
Gaps (required for fire damper thermal expansion) around fire dampers
Other instrument lines
Previous repairs with RTV sealant
Through bolts for hangers or equipment

Basic guidelines for inspection are as follows, however, specific requirements will vary with application, equipment vendor, type of sealant, etc. The term "approved," as used below, means that the material, component or technique has been approved by the plant engineering staff for the particular application.

3.1 Doors and Door Seals

The door should fit properly in the frame, with hinges securely attached. Door sweep should be in continuous contact with the floor or threshold for the entire width of the door. The gasket or seal should be an approved type, be free of cracks and should form a contact seal around the entire perimeter of the door. The door and frame should be free of breaks or open holes. With the door closed, the seal should be compressed against the door at all points.

3.2 Dampers

Dampers, associated linkages and actuators should be inspected for proper movement throughout the entire range of travel. If applicable, response to actuation signals and required cycle time should be verified. Commensurate with the design and safety analysis requirements, seal tightness should be verified. Frames should be checked for dimensional stability and be structurally sound. Frame-to-wall gaps should be minimized and consistent with vendor and UL requirements. Damper gaskets or seals, if required, should be an approved type, be free of cracks and should form a contact seal around the entire perimeter of the damper or where installed. The damper and frame should be free of breaks or open holes. With the damper closed, the seal should be evenly compressed against the damper at all points.

3.3 Gaps At Building Wall/Floor/Ceiling Intersections

All walls and intersections of the CRE should be visually inspected for integrity. Deficiencies in original construction, building differential settlement and deterioration of sealing materials can result in significant but unnoticed openings in the CRE. Due to equipment, cabling and other interferences, these areas are difficult to inspect. Repairs should be made using approved sealants or grouts, in accordance with vendor instructions.

3.4 Ducting, Duct Penetrations, Expansion Joints

Welded ducting is preferable for CR HVAC ducting outside the CRE and for other ductwork running through the CRE. For other types, all seams and connections should be sealed with an approved sealant, such as room temperature vulcanization or hardcast, and tested for leak tightness (Snoop or pressure decay methods). Duct penetrations should also be sealed with an approved sealant or grout.

Expansion joints should be sealed and firmly clamped at each end, and should be free of cracks, holes and or tears. If replacement of the joint is necessary, old adhesive should be removed from the mating surfaces should be inspected for defects. The length and width of the joint should allow for at least a one-inch overlap at each end. If the duct is located outside, additional width should be included for slack, and the material should be rated for sun and weather exposure, or be covered with an approved coating.

3.5 Electrical Cables, Conduits, Cable Trays

All electrical conduits and cable trays penetrating the CRE should be sealed with an approved sealant. Sealing of the inside of the conduits is especially important due to the large potential flow areas that may not be readily apparent during a normal visual walkdown or inspection.

Close attention should be paid to the condition of penetrations. Typically, many wall and floor penetrations are sealed with a silicone foam. Although the penetration may appear to be sealed, in-leakage may still be occurring due to shrinkage of the foam, voids in the seal due to cable relaxation, voids between the cables in cable bundles and improper cure of the foam. Delamination of material in wall seals is also possible.

Electrical conduits and cable trays provide a significant potential source of in-leakage due to the large number of these components. Normal problem areas include unsealed conduits that terminate inside the CRE, intermediate connectors, junction boxes and panels, and non-leak-tight flexible conduit. Cable trays that are not filled completely by cable may leave voids that may have been overlooked during initial construction and sealing efforts.

3.6 Instrumentation Or Air Tubing

All instrumentation or air tubing penetrating the CRE should be inspected for potential leakpaths such as open valves in abandoned lines or insufficient seal around the tubing.

3.7 Air Handling Unit (AHU) / Fan Housings And Shafts

Inlet and outlet flanges should be sealed with approved sealants, or preferably continuously welded on both sides. Any fan housing drains should have plugs installed. AHU drain loop seals should be verified periodically. Separate sections of AHU housings should have individual drains. High quality or double gaskets (not sealants) should be used on cover plates and access doors. Bolts on cover plates and access doors should be spaced on 3" to 4" centers. Recommended shaft seals are stuffing box seals, lip seals or mechanical type seals. An arrangement using a neutral purge gas is also effective.

3.8 Plumbing Equipment

All plumbing-related equipment in the CRE should be checked for potential leak paths. Floors, restrooms, kitchens, showers and water fountains have drains. These drains must have traps and should be inspected regularly to verify they are filled. Abandoned traps and piping should be permanently closed or sealed.

4. Alternatives to Sealing

As indicated above, there are many opportunities for degradation of the CRE to occur, such as normal equipment wear and changing operational practices. It may be advantageous, therefore, to consider alternatives to supplement the sealing program.

- Problem: Major equipment (AHUs, filters, dampers, etc.) and long duct runs located outside the envelope significantly increase the potential for unfiltered in-leakage, and the effort required to detect and measure the in-leakage.
- Solution: Permanently moving this equipment or ducting inside the envelope by expanding the boundary walls, floors, etc, may be a cost-effective means of reducing this problem.
- Problem: Airflow balance inside the CRE may produce unfavorable pressure differentials within separate spaces in the CRE, leading to potential positive pressure differentials relative to the outside or adjacent spaces.
- Solution: Careful flow balance testing may be required to resolve this problem. Maintaining CRE internal doors open, adding door louvers to internal doors or installing additional supply/return registers can improve pressure communication within the CRE and prevent this problem.

- The design and operation of ventilation systems serving adjacent spaces, safety-related as well as non safety-related, should be reviewed to prevent unfavorable CRE-adjacent space pressure differentials post accident.
  - This evaluation should consider scenarios both with and without off-site power.
  - From a CRE perspective, an accident without a loss of off-site power (LOOP) may actually be worse due to continued operation of non-safety ventilation systems in adjacent spaces. In some cases, modifications should be considered to shut off non-safety exhaust or supply fans in the event that a LOOP does not occur.

APPENDIX K

CONTROL ROOM ENVELOPE BOUNDARY CONTROL PROGRAM

1. Purpose/Scope

This appendix provides guidance to control breaches of the CRE and may be used to develop plant specific procedures.

2. Scope

A boundary control program should manage activities that breach the CRE such as:

- The creation of a new penetration in the CRE
- Opening of an existing penetration in the CRE
- Any activity that restricts the normal closure of a CRE door
- The removal of a CRE door/hatch from its design location
- The blockage or breach of a CRE ventilation duct
- Removal of or changes to structural components such that CRE boundary leak tightness may be affected
- Removal of fire, steam, high energy line break or flood barriers that also serve as the CRE boundary
- Any piping system breach (e.g., valves, pumps or pipes) that creates an air flow path through the CRE boundary.
- The removal of equipment and/or floor drain plugs from the CRE boundary.

3. Discussion

The physical CRE boundary is a fundamental element of CRE integrity. It is important to control the CRE boundary to ensure that the design is maintained such that the accident analyses remain valid. In the event that planned maintenance, testing or plant conditions have potential to affect the CRE boundary, administrative control of the boundary should be procedurally maintained. This includes controlling openings in the boundary required for maintenance and modifications as well as preventing inadvertent openings. A program should be in place to:

- Evaluate the impact on the accident analyses when breaching the boundary
- Monitor active breaches
- Ensure pre-planned responses to close the breach in the event of a toxic gas or radiological challenge are in place
- Ensure that the boundary is restored.

Baseline testing measured the actual CRE in-leakage. This measured value is typically less than the maximum in-leakage that can be tolerated and still meet regulatory limits. The difference between these two values establishes a margin that can be used to determine the maximum allowable size of a CRE breach to ensure that system operability is maintained.

The breach size can impact the ability of a positive pressure control room to maintain the minimum required differential pressure across the CRE boundary. If positive pressure cannot be maintained, this may result in greater in-leakage. Additionally, the maximum pressurization airflow rate allowed by the accident analyses may be adversely affected.

4. PROCESS

4.1 Impact Evaluation

Prior to breaching the CRE boundary, the activity should be evaluated for the impact on control room habitability. This evaluation should consider, as a minimum, the breach size and the ability to maintain the CRE integrity or rapidly restore the boundary. The impact on fire boundaries, tornado protection boundaries, security boundaries, etc., should also be considered when opening up a boundary.

4.2 Breach Size

Evaluate the effect the breach has on in-leakage margin, pressurization flow rate and required differential pressure across the boundary.

The first step in determining the maximum breach size is to identify the allowable in-leakage based on the margin of the accident analyses. The second step is to determine the impact on the differential pressure across the boundary that will be breached under accident conditions. The third step is to calculate the maximum breach size using the allowable in-leakage and differential pressure as inputs to the orifice equation. If the anticipated breach size is less than the maximum breach size, the activity is allowed.

For positive pressure control rooms, a test should be performed to verify that the breach size does not adversely impact the CRE differential pressure and pressurization air flow requirements.

If it can be demonstrated that the breach will not be open long enough to result in exceeding toxic gas or dose limits, then the maximum breach size does not need to be calculated.

If the breach size adversely impacts the accident analyses or system performance requirements, compensatory measures may be necessary. These compensatory measures may need a 10 CFR 50.59 evaluation.

4.3 Ability to Rapidly Restore the Boundary

Breaches such as blocking doors open do not require evaluation if the breach can be quickly restored. To make use of this exception, a person must be assigned whose primary responsibility is to shut the door at the onset of abnormal conditions. The assigned individual must also be in communication with the control room.

4.4 Breach Monitoring

At any given time, multiple breach activities may be in progress. Controls should be in place to monitor the number of breaches and ensure that the sum effect of all the active breaches does not result in exceeding regulatory limits. This may be accomplished via a breach permit tracking system, differential pressure monitoring or controlling the number of work orders that impact control room habitability.

4.5 Boundary Restoration

The breach shall be verified closed when the barrier has been restored (e.g., qualified penetration seal installed) and work-related compensatory measures removed. All restoration activities should be documented.

APPENDIX L

GLOSSARY OF TERMS

1. Purpose / Scope

This appendix contains abbreviations, acronyms, and definitions applicable to the entire document.

2. Abbreviations and Acronyms

ACRS	Advisory Committee on Reactor Safeguards
ADV	Atmospheric dump value
AEC	Atomic Energy Commission (U.S.)
AFW	Auxiliary feedwater
AHU	Air handling unit
ANSI	American National Standards Institute
AST	Alternative source term
ASTM	American Society for Testing and Materials
ATWS	Anticipated transient without SCRAM
BBP	Barrier breach permit
BWR	Boiling water reactor
CEA	Control Element Assembly
CFR	Code of Federal Regulations
COLR	Core operating limit report
CPR	Critical power ratio

CR	Control room
CRE	Control room envelope
CREVS	Control room emergency ventilation system
CRH	Control room habitability
DBA	Design basis accident
DCF	Dose conversion factor
DEQ	Dose equivalent
DEI	Dose equivalent iodine
DF	Decontamination factor
DNB	Departure from nucleate boiling
DOE	Department of Energy (U.S.)
ECCS	Emergency core cooling systems
EDG	Emergency diesel generator
EDO	Executive director of operations
EAB	Exclusion area boundary
EOP	Emergency operating procedure
EPCRA	Emergency Planning and Community Right-to-Know Act
ERDA	Emergency response data alarm
ESF	Engineered safety feature
FHA	Fuel handling accident
FSAR	Final safety analysis report
GDC	General design criteria(on)

GL	Generic letter
GSI	Generic safety issue
HELB	High-energy line break
HFP	Hot full power
HPSI	High pressure safety injection
HVAC	Heating, ventilation and air conditioning
ICRP	International Commission on Radiological Protection
IDLH	Immediately dangerous to life and health
IEN	Inspection and enforcement notice
IN	Information notice
INPO	Institute of Nuclear Power Operations
IPF	Iodine partition factor
JCO	Justification for continued operation
KI	Potassium Iodine tablet
LCO	Limiting condition for operation
LEPC	Local emergency planning committee
LOAC	Loss of AC
LOCA	Loss-of-coolant accident
LOOP	Loss of off-site power
LPZ	Low population zone
LRA	Locked rotor Accident
LWR	Light water reactor

MHA	Maximum hypothetical accident
MSLB	Main steam line break
MSIV	Main steam isolation valve
MSIVLCS	MSIV leakage control system
MTC	Moderator temperature coefficient
M&TE	Measuring and test equipment
NEI	Nuclear Energy Institute
NHUG	Nuclear HVAC Utilities Issues Group
NIOSH	National Institute for Occupational Safety and Health
NOV	Notice of violation
NRC	Nuclear Regulatory Commission (U.S.)
NRR	Nuclear Reactor Regulation
OIE	Office of Inspection & Enforcement
PMT	Post-maintenance test
PNL	Pacific Northwest Laboratories
PORV	Power operator relief valve
PSAR	Preliminary safety analysis report
PWR	Pressurized water reactor
RAI	Request for additional information
RCP	Reactor coolant pump
RCS	Reactor coolant system
REA	Rod ejection accident

REM	Roentgen equivalent man
RHR	Residual heat removal
RMP	Risk management program
RTV	Room temperature vulcanization-as used in this document commonly refers to sealants containing silicon that cure at room temperature.
RWA	Rod withdrawal accident
RG	Regulatory guide
SBO	Station blackout
SCBA	Self-contained breathing apparatus
SER	Safety evaluation report
SERC	State emergency response commission
SG	Steam generator
SGTR	Steam generator tube rupture
SGTS	Standby gas treatment system
SI	Safety injection
SIAS	Safeguards initiation activation signal
SMACNA	Sheet Metal and Air Conditioning Contractors National Association
SRP	Standard review plan
SSC	Structure, system or component
SSE	Safe shutdown earthquake
TEDE	Total effective dose equivalent
TMI	Three Mile Island

UFSAR	Updated Final Safety Analysis Report
χ/Q	Atmospheric dispersion coefficient typically pronounced "chi over q"

3. Definitions

AIR CHANGE FLOW (from ASTM E741): The total volume of air passing through the zone to and from the outdoors per unit of time.

AIR CHANGE RATE (from ASTM E741): The ratio of the total volume of air passing through the zone to and from the outdoors per unit of time to the volume of the zone.

ATTENDANT: The individual assigned to carry out the compensatory actions defined in the barrier breach permit.

BOILING WATER REACTOR: A reactor in which water, used as both coolant and moderator, is allowed to boil in the core. The resulting steam can be used directly to drive a turbine and electrical generator, thereby producing electricity.

BOUNDARY: A combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE.

BREACH: Any work activity or testing that creates or enlarges an opening through a barrier, which would allow the propagation of a hazard through the barrier.

- Modification (addition, removal or degradation) of a penetration seal or structural component
- Core boring
- Blocking open a door/hatch or damper
- Modification (addition, removal, or degradation) of a door/hatch or damper

DEPARTURE FROM NUCLEATE BOILING: The point at which the heat transfer from a fuel rod rapidly decreases due to the insulating effect of a steam blanket that forms on the rod surface when the temperature continues to increase.

DESIGN BASES: Information that identifies the specific functions to be performed by a structure, system or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state-of-the-art" practices for achieving functional goals or (2) requirements derived from analysis (based on calculations and/or experiments) of the effects of a postulated accident for which a structure, system or component must meet its functional goals. (10CFR50.2)

DESIGN BASIS ACCIDENT: A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures and components necessary to assure public health and safety.

EMERGENCY CORE COOLING SYSTEMS: Reactor system components (pumps, valves, heat exchangers, tanks and piping) that are specifically designed to remove residual heat from the reactor fuel rods should the normal core cooling system (reactor coolant system) fail.

ENGINEERED SAFETY FEATURE: Features that are designed to automatically start to mitigate the consequences at an accident. These systems help to prevent and/or control the release of radiation.

FILTERED IN-LEAKAGE: This is leakage that occurs at a location that allows contamination to be filtered prior to the air entering the habitability zone. An example is duct leakage on the suction side of a pressurization filter system where the duct is outside the control room envelope. Radionuclides are removed from this air prior to it entering the habitability zone. There is no filtering assumed for toxic gas events.

GAP: The space inside a reactor fuel rod that exists between the fuel pellet and the fuel rod cladding.

HAZARD: A condition or event that could jeopardize the operation of risk significant equipment. Examples are fire, water, air, steam, smoke, CO_2, toxic gas, hot gas and security.

HAZARD BARRIER: A wall, floor/ceiling, penetration, door or hatch constructed of building materials used to physically separate areas and contain hazards.

HAZARD DOOR/HATCH: barriers used to physically separate areas and contain hazards. Examples are doors, blowout panels, dampers, or hatch plugs.

INOPERABLE BARRIER: A barrier that is inoperable such that it can not fully perform its intended function.

INTEGRATED TRACER GAS TEST: A tracer gas test to determine total leakage of the CRE. The tracer gas test is actually measuring the amount of air changing in the space (i.e., the air going out is being replaced by the air going in). This particular test does not locate leaks; it only provides a value for total in-leakage.

LICENSING BASIS IN-LEAKAGE: This is the in-leakage that is used in the plant design basis radiological analysis with design basis values of other plant parameters to calculate control room operator dose during a licensing basis accident.

LIMITING CONDITION FOR OPERATION: The section of Technical Specifications that identifies the lowest functional capability or performance level of equipment required for safe operation of the facility.

LOSS-OF-COOLANT ACCIDENT: Those postulated accidents that result in a loss of reactor coolant at a rate in excess of the capability of the reactor makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.

MAXIMUM ALLOWABLE RADIATION IN-LEAKAGE: This is the value assumed in the current licensing basis analysis. Calculated in-leakage value in cfm that will result in the control room operators receiving the maximum allowable dose with design basis inputs of all other parameters to the plant radiological analysis. This value must be calculated for each plant.

MAXIMUM ALLOWABLE RADIATION IN-LEAKAGE FOR OPERABILITY DETERMINATION: This is the calculated in-leakage value in cfm that will result in the control room operators receiving the maximum allowable dose with realistic but verifiable inputs of all other parameters to the plant radiological analysis. This value may take credit for compensatory measures allowed by GL 91-18.

MAXIMUM ALLOWABLE TOXIC GAS IN-LEAKAGE: This is the maximum calculated in-leakage of toxic gas that will result in the control room remaining habitable for the bounding toxic gas hazard evaluation.

MAXIMUM HYPOTHETICAL ACCIDENT: The design basis accident that results in the largest off-site dose.

MOX: Mixed oxide nuclear fuel

NDT: Non-destructive testing

PENETRATION: An opening in a CRE boundary wall or floor/ceiling, other than a door/hatch, which contains materials or mechanical devices that prevent the propagation of a hazard through the barrier. Some examples are:

- Penetration seals
- Structural material
- Dampers for example, fire, tornado, etc.

PRESSURIZED WATER REACTOR: A power reactor in which heat is transferred from the core to an exchanger by high temperature water kept under high pressure in the primary system. Steam is generated in a secondary circuit. Many reactors producing electric power are pressurized water reactors.

RESPONSIBLE ENGINEER: Designated engineer for hazard barrier programmatic controls.

ROENTGEN EQUIVALENT MAN: A standard unit that measures the effects of ionizing radiation on humans.

SAFE SHUTDOWN EARTHQUAKE: A design-basis earthquake.

SAFETY EVALUATION REPORT: A report attached to an NRC approval letter that documents the NRC's basis for approval of a submittal.

SAFETY INJECTION: The rapid insertion of a chemically soluble neutron poison (such as boric acid) into the reactor coolant system to ensure reactor shutdown.

STATION BLACKOUT: Loss of all AC power at a single unit plant or on one of the units at a dual unit plant.

TID: Technical information document

TRACER GAS (from ASTM E741): A gas that can be mixed with air in very small concentrations in order to study air movement.

UNFILTERED IN-LEAKAGE: This is leakage that occurs at a location in the habitability system that allows air to enter the control room envelope without any contaminants being removed at the point of entry. Examples would be penetrations and dampers that are at a negative pressure with respect to potentially contaminated surroundings and located such that radionuclides are not removed prior to the in-leakage entering the control room.