NEI 99-03 [Draft]

Control Room Habitability Assessment Guidance

October 2000

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Nuclear Energy Institute

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October 2000

ACKNOWLEDGEMENTS

To be developed

NOTICE

EXECUTIVE SUMMARY

To be developed

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APPENDIX

PART 1 - BACKGROUND

1 INTRODUCTION

1.1 PURPOSE

This document provides guidance on how to demonstrate adequate protection of control room operators against the effects of postulated external releases of radioactivity or toxic gases. It also provides guidance on the development of a Control Room Integrity Program to facilitate long-term maintenance of the control room envelope. This document presents a process to demonstrate that the licensing and design bases associated with control room habitability (CRH) are satisfied.

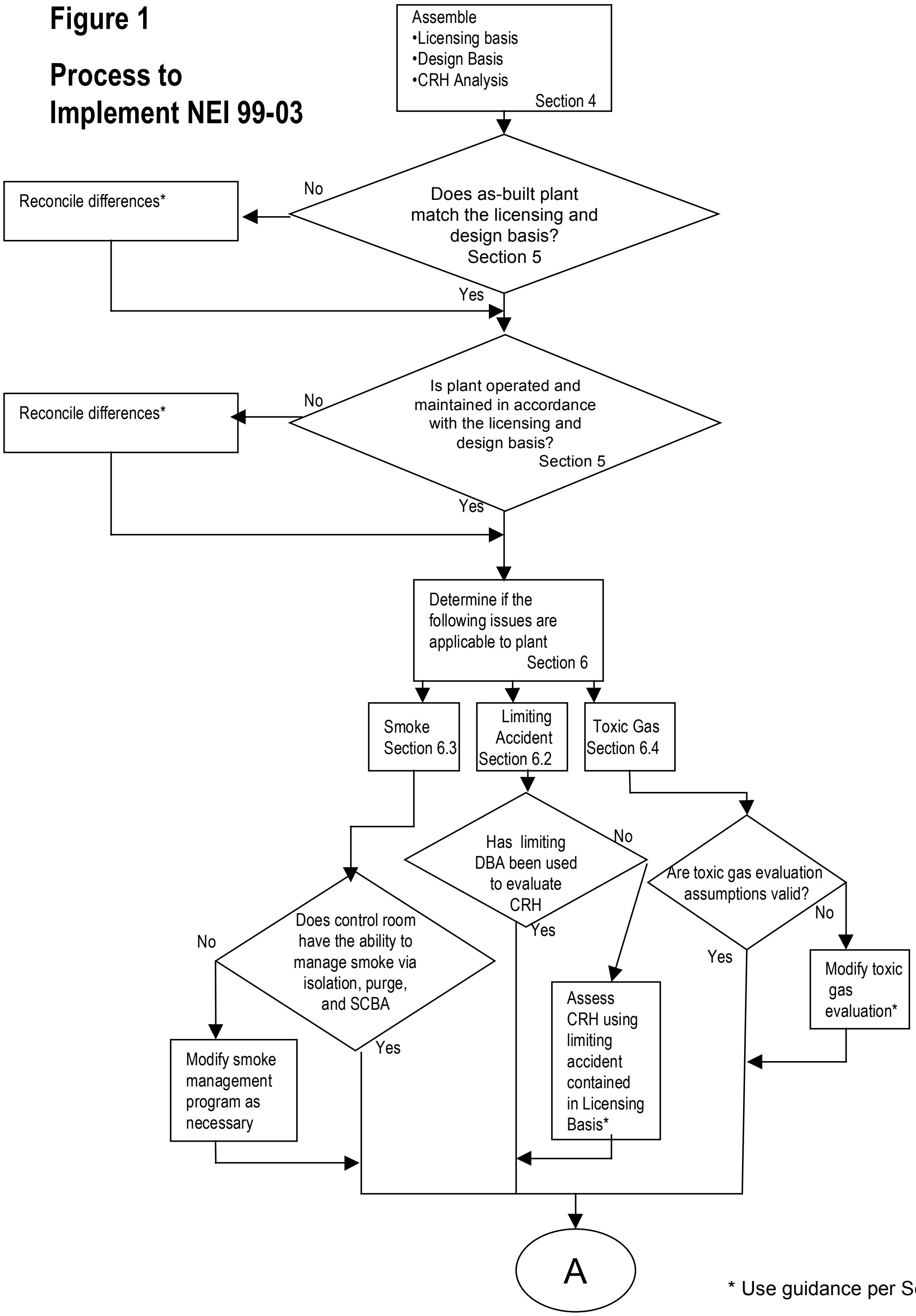
Use of this document is voluntary and defines an acceptable method to establish and maintain CRH.

1.2 SCOPE

This guidance document demonstrates adequate protection of the control room operators within the limits of the plant's existing design and licensing bases. Figure 1 demonstrates the process used to achieve this goal. Once adequate protection capabilities have been demonstrated, the document provides options to aid in developing credible programs to maintain control room integrity and habitability.

Figure 1

Process to Implement NEI 99-03



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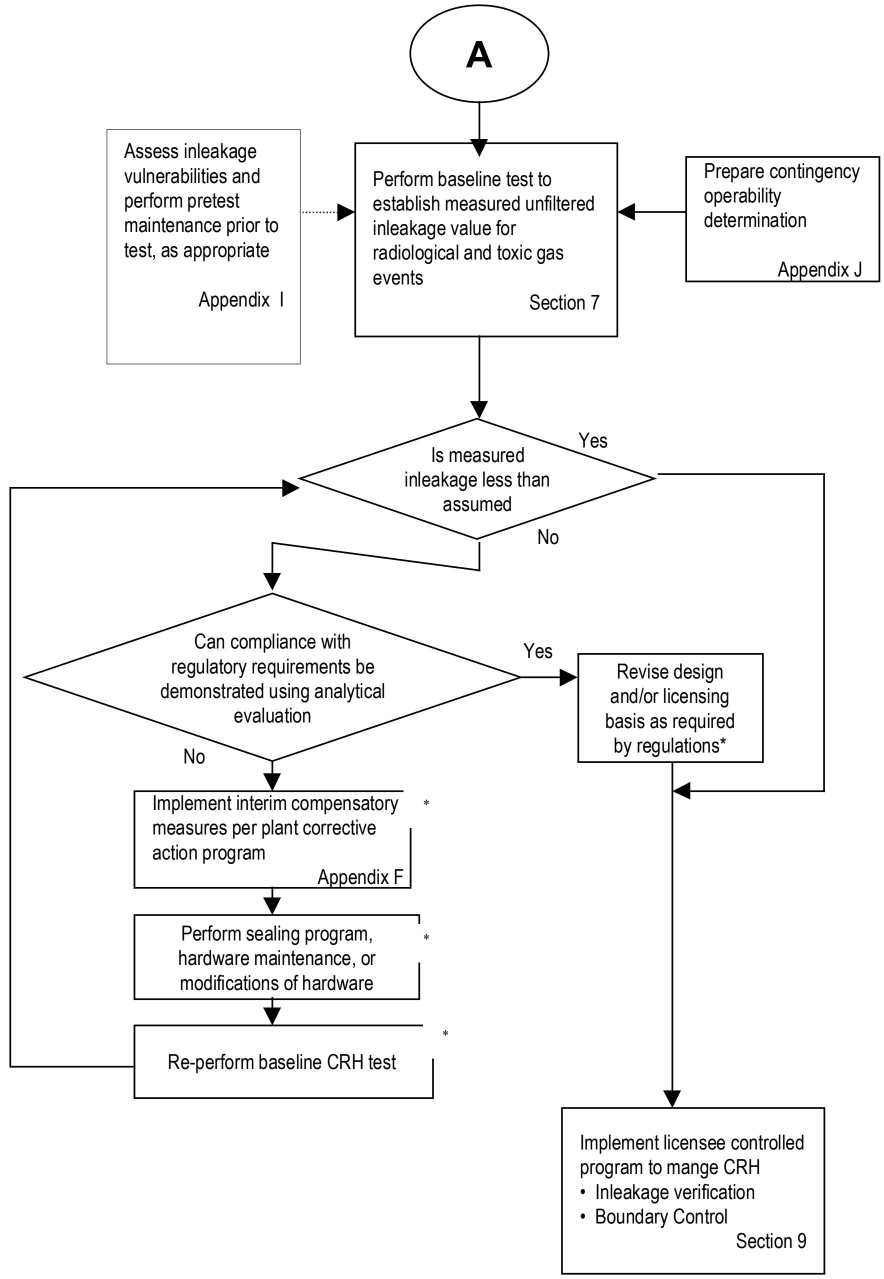

* Use guidance per Section 8

1.3 HISTORY

In 1971, 10CFR50, Appendix A, General Design Criterion 19 (GDC 19) became regulation. GDC 19 required 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 were mandated by the Nuclear Regulatory Commission (NRC) for licensees to evaluate their control rooms to assure adequate protection of operators. An individual licensee's degree of compliance with GDC 19 and/or commitment to the TMI Action Plan can be found in the plant specific licensing basis. The development of a plant's licensing basis was dependent upon the status of the nuclear plant at the time of issuance of its operating license and the regulatory documents.

Since the mid-1980s, the NRC has held meetings and issued information notices concerning inadequacies of control room designs in assuring that CRH requirements were met. Between 1980 and 1996, numerous documents were published to assist the industry in evaluating control rooms for effects of toxic gas and radiation. In the mid-1990s, testing of some licensee control room envelopes indicated that key assumptions supporting the radiological dose analysis may be incorrect such that the ability to meet regulatory requirements is suspect. In 1998, the NRC held a public workshop to express concerns similar to those from the mid-1980s. In late 1999, the NRC and the industry agreed to work together on issues affecting CRH and develop this guidance document for resolving those issues.

See Appendix A for more information.

1.4 DOCUMENT ORGANIZATION

This document defines a process for licensees to assess a plant's design and licensing bases to assure that they remain appropriately maintained throughout the life of the plantAppendices are included that provide detailed guidance for completing the assessment.

Figure 1 provides the process flow used in this guidance document.

This document is divided into three parts:

- Background
- Assessment Process
- Establishing And Maintaining Control Room Envelope (CRE) Integrity

Part 1, *Background*, provides information necessary to perform the assessment and 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 a process to determine if the plant configuration and operation is 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, *Industry Issue Applicability*
- Section 7, *Air Inleakage*
- Section 8, *Methodology for Dispositioning and Managing Discrepancies*

The first step for the assessment is discussed in Section 4, *Determining CRH Licensing Basis.* This section provides guidance on identifying and assembling the current plant licensing and design bases. This information will be compared with actual plant configuration and operation. Section 5, *Comparing Existing Plant Configuration and Operations with Licensing Bases for CRH,* provides guidance for determining if the plant configuration and procedures have remained aligned with the licensing basis.

Section 6, *Industry Issue Applicability,* provides guidance for determining the applicability of the Section 3 industry issues except for inleakage, which is the subject of Section 7. Section 6 provides guidance for the licensee to determine if its plant is susceptible to the industry issues and identifies appropriate actions.

Section 7, *Air Inleakage*, recommends the performance of a baseline test to determine the amount of unfiltered inleakage. The test may be performed using the ASTM E741 tracer gas methodology or a component test methodology. The purpose of this baseline test is to determine if the inleakage is consistent with that used in the CRH evaluation. Section 8, *Methodology for 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, *Establishing and Maintaining CRH*, provides guidance on implementing a licensee controlled program to manage CRH after the evaluations of Part 2 are completed. It is composed of:

- Section 9, *Long-Term CRH Program*

Section 9 provides guidance on establishing a licensee CRH program that includes periodic evaluation of inleakage and maintenance of CRE integrity.

2 REGULATORY REQUIREMENTS AND GUIDANCE

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

2.1 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 may have 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.

The regulation provides acceptance criteria for only the radiation protection function. For most licensees, the dose acceptance criteria of 5 rem whole body to an individual in the control room should not be exceeded for any postulated design basis accident. Standard review plan interprets this requirement to be satisfied by a thyroid or a beta skin limit of 30 rem. With the issuance of the Alternative Source Term (AST) Rule (10CFR50.67), in December 1999, the dose acceptance criterion was established at 5 rem TEDE for licensees implementing the AST. GDC 19 requires, however, a control room "from which actions can be taken … under accident conditions." Acceptance criteria for non-radiological accidents (i.e., toxic gas release) are provided in other guidance documents discussed below.

2.2 REGULATORY GUIDANCE

This subsection provides an overview of various types of documents that supply regulatory guidance relative to CRH. As the documents discussed below 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.

2.2.1 Regulatory Guides

Regulatory Guides provide one acceptable approach for satisfying regulations. Use of a regulatory guide is voluntary. There are 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.

2.2.2 NUREGs

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

In particular, NUREG-0737 action item III.D.3.4, deals directly with CRH. It directs applicable plants to re-confirm compliance with GDC 19. It, and the plant response, may be key considerations as a plant researches its licensing basis.

The other NUREGs described in the appendix provide guidance on analysis assumptions, modeling, and system vulnerabilities.

2.2.3 Inspection and Enforcement Notices (IEN)

These documents were issued to inform licensees about events, issues, and generic observations. The documents did not require a response and so plants generally do not have docketed commitments to incorporate changes into the design or operation practices as a result of IENs.

Applicable IENs are identified in Appendix B to provide information on plant operating experience with designs or events with an impact on CRH.

2.3 GENERIC ISSUES

There have been two generic safety issues (GSI), B-66 and 83, related to CRH. Information on these GSIs, is located in NUREG-0933, which is summarized in Appendix B. This information provides insight into the historical evolution of the CRH concerns.

3 INDUSTRY ISSUES ASSOCIATED WITH CONTROL ROOM HABITABILITY

3.1 SCOPE/PURPOSE

At a July 1998 workshop, the NRC and industry discussed issues relating to control room habitability (CRH). This section summarizes the following issues:

- Licensing basis different from as-built plant
- Analyses different from as-built or as-operated
- DBA analyzed not most limiting
- Smoke infiltration
- Toxic gas evaluation
- Control room inleakage greater than assumed

3.2 LICENSING BASIS DIFFERENT FROM AS-BUILT PLANT

During review of license amendments, licensees and the NRC staff have observed that some plants have introduced inconsistencies between the plant's licensing basis and the as-built plant. Anecdotal evidence of differences between the description of the control room envelope and the HVAC systems controlling the air flow within this envelope and the as-built condition of the plant have been noted. 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.

3.3 ANALYSES DIFFERENT FROM AS-BUILT OR AS-OPERATED

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 be 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 not subject to review during the original licensing phase, that could impact the results of a licensee’s CRH analysis. Licensees should assess the impact of these changes on CRH and the supporting analyses.

3.4 DBA Analyzed Not Most Limiting

Each plant is required to analyze the limiting design basis accident (DBA) 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 this is not always true.

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 a large amount of inleakage from outside the envelope. The concern is that smoke in the control room could challenge the ability of the operator to remain on station to shut down the reactor. No explicit 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.

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 must be evaluated.

3.7 Control Room Inleakage Greater Than Assumed

Tracer gas tests have been conducted at over 15 nuclear power plant control rooms to determine the amount of inleakage (filtered and unfiltered). Test results to date indicate that measured inleakage is greater than the amount assumed in CRH design basis analyses. In some cases the increase was very significant. This is a concern because the control room inleakage value is an input to the evaluation of both radiological and toxic gas events.

3.7.1 Radiological Considerations

The primary concern is that increased control room unfiltered inleakage could result in the reactor operator being exposed to a larger dose than previously analyzed. Unfiltered inleakage rates are one of several inputs into the analyses used to determine operator doses. The term *unfiltered* refers to air leaking into the control room envelope[1] that does not pass through either a charcoal filter or HEPA filter. With higher unfiltered inleakage, the iodine removal credited in the accident analyses may be inaccurate and non-conservative.

3.7.2 Toxic Gas Considerations

Inleakage is also a concern for toxic gas events. The amount of inleakage 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 inleakage to be used in the toxic gas analysis.

[1] The area containing plant controls necessary for safe plant operation and occupied by nuclear plant operators. This may include, as applicable, locker rooms, office spaces, lavatories, equipment rooms, etc. This is also known as the habitability zone and is served by the control room emergency ventilation system.

PART 2 – ASSESSMENT PROCESS

4 DETERMINING CRH LICENSING BASIS

4.1 PURPOSE/SCOPE

This section provides information, references and guidance that will help the licensee identify its Control Room Habitability (CRH) licensing basis. The information identified will be used though out the remainder of this guidance document.

4.2 UNDERSTANDING THE CONCEPT OF LICENSING BASIS

Licensees must know the CRH licensing basis in order to apply the process defined in this document. One goal of the guidance is for licensees to compare the existing licensing basis to existing conditions in order to confirm that CRH has been established and is being maintained. In addition, knowledge of the CRH licensing basis will also assist licensees in:

- Making informed decisions regarding proposed changes to the physical plant,
- Responding to questions from the regulator when license amendments are proposed, and
- Dispositioning corrective actions for degraded conditions.
- 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 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 ECCS, SBO and ATWS rules.

4.2.2 Supporting Design Information

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

4.2.3 Licensing Basis

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

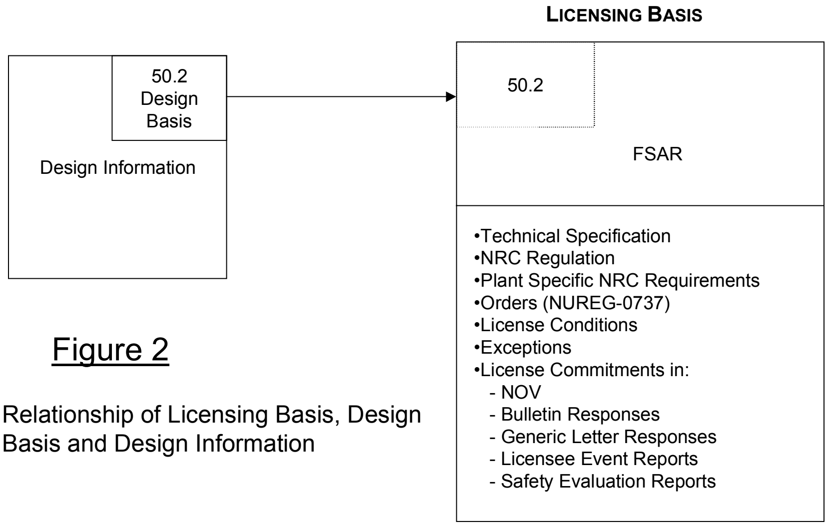

Figure 2

Relationship of Licensing Basis, Design Basis and Design Information

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 a licensee 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 that their plant was licensed or other criteria they committed to.

The NRC SERs document the boundaries of the licensing action proposed by the applicant, the applicant's analysis, a staff 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. An SER cannot establish a commitment binding on the licensee. The licensing basis is defined by the information submitted by the licensee.

The important point is that a plant's licensing basis consists of only those items that it is required by law to meet or to which it has committed itself.

4.3 LICENSING BASIS SOURCES

Licensees document compliance with regulatory requirements in various documents. Specific documents to include in a review are the:

- UFSAR

- Licensing correspondence that contain commitments and exceptions to applicable regulatory requirements and regulatory guidance
- Operating license and amendments
- Technical Specifications and their bases
- 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 information, the process identified in NEI 97-04 is useful for assembling the plants licensing basis.

4.5 Assembling The CRH Analysis

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.

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

The implementation of a CRH licensing basis identification program will identify open items that may include questions, concerns, and cases of missing information. Other items that 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 this part of the process is provided in following sections.

A licensee should know the plant's current design basis for CRH mitigation features. This includes:

- Design basis accidents within the plant's licensing basis
- Components that provide a radiological, toxic gas, or smoke mitigation function and their specific performance requirements

- Analysis inputs, such as the amount of unfiltered inleakage, their bases, and source documents. For example, inputs such as occupancy factors may have been adopted from the Standard Review Plan. 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 the most limiting accident is identified.
- All modes of control room ventilation system operation and system alignments necessary to mitigate radiological and toxic gas events and 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
- The plant's current licensing basis for CRH. Including determination of the plant's :
    - Technical Specification performance limits and surveillance requirements for credited components
    - Commitments regarding operation of the control room envelope
    - Other requirements regarding operation of the control room envelope, which may be identified in such documents as the licensee's SAR, Design Criteria Memoranda, operating procedures, surveillance test procedures, etc.
    - Submittals involving amendments associated with steam generator replacement, steam generator alternate repair criteria and power uprates with assumed criteria for evaluating the effect on CRH.

4.6 DOCUMENTATION OF THE EXISTING PLANT CRH LICENSING AND DESIGN BASIS

The licensee should document the plant CRH licensing and design basis review just performed.

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

5.1 PURPOSE

After the licensing and design bases have been compiled, an assessment of the system configuration, operation, and maintenance should be performed.

This comparison is needed because after years of operation many new procedures and methods of operation, maintenance and testing have been developed and revised. Systems may be operated differently from the assumptions or values used in analyses that determine operator exposure from radiological or toxic gas events. Given the dynamic nature of the 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

The as-built configuration should be reviewed to ensure that the construction and configuration satisfies the design and licensing bases. For example:

- Plant drawings should be reviewed to ensure that the design would provide the desired function and support the inleakage 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 desired leak tightness?
- A system walk down should be performed to ensure that the actual field configuration agrees with the plant drawings/design.

5.3 REVIEW THE NORMAL AND EMERGENCY OPERATING PROCEDURES (EOPS) AFFECTING THE CONTROL ROOM VENTILATION SYSTEMS

Plant operating procedures should be reviewed 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.
- Normal operating procedures should align the system to ensure that the proper flow paths are established. Damper settings should be correct to establish the proper flow rates.
- Ensure the EOPs verify/place the control room ventilation system in the proper configuration/alignment 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 that they are consistent with the following:

- The procedure should adequately demonstrate 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.

5.5 REVIEW THE MAINTENANCE PROGRAMS AND PROCEDURES FOR SENSITIVITY TO CRH REQUIREMENTS

Assess maintenance activities 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.
- Structural Maintenance near the control room should be reviewed to ensure that the envelope is not inadvertently breached.
- Maintenance procedures for system components should address system integrity requirements. It should be noted that removal of inspection plates or opening access doors might constitute a breach of the CRH 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 address likely sources of control room inleakage. 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 design control procedures to ensure that changes that may have a direct or indirect impact on CRH are properly evaluated. 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 inleakage values.
- Electrical work such as installing new conduit or pulling cable could create new inleakage paths.
- Installing floor or equipment drains can result in unexpected inleakage paths

5.7 Review the CRH Analyses

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

- Do the system lineups assumed in the CRH analyses agree with the current procedures?
- Are the 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?
- Are the assumptions in the toxic gas analysis valid?

5.8 Identified Inconsistencies

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

6 INDUSTRY ISSUE APPLICABILITY

6.1 SCOPE

This section provides guidance for evaluating the applicability of industry issues to specific plants. It also recommends actions to address applicable industry issues. Section 3 provides additional information on these issues. This guidance follows the approach outlined in Figure 1.

The industry issues discussed in this section are:

- Limiting Design Basis Accident (DBA)
- Smoke infiltration
- Toxic gas evaluation

Another important industry issue, inleakage, is discussed in Section 7.

6.2 LIMITING DBA

The loss of coolant accident (LOCA) DBA was frequently assumed by licensees to be the bounding accident. Licensees frequently used the accident to assess the adequacy of CRH design. They may not have adequately considered the impact of different plant responses or atmospheric dispersion for other accidents on radiological consequences to the reactor operators. These other accidents may include accidents at adjacent units.

6.2.1 Recommended Actions to Evaluate Limiting Accidents

The licensee should examine each DBA listed in the FSAR for which offsite doses have been reported to determine the event that is limiting with respect to control room dose. FSAR-described accident scenarios, postulated source terms and assumptions, combined with a control room model using appropriate system parameters and responses as well as atmospheric dispersion factors, should be used. If the facility license or FSAR requires consideration of releases from accidents at an adjacent plant, 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 which accident is limiting with respect to CRH include:

- For accidents where the CRH features are actuated by containment isolation or safety injection (SI) signals[2], 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. Licensees that evaluated LOCA as the limiting DBA may not have adequately considered the impact of this delay on radiological consequences to the reactor operators.
- 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 play a role as to 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 downwind of the control room intake, while those for other accidents may 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 plants with a secondary containment or enclosure building.
- For PWRs that have approved alternative repair criteria for steam generators, the MSLB accident is generally the limiting accident with regard to control room habitability as such facilities have maximized the postulated dose in order to maximize the repair criteria.

6.2.2 Adjacent Unit Accidents (a special case)

A special case of limiting DBA could be the presence of an accident release from an adjacent plant. 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 required by the license for the plant evaluating its control room..

In addition, if there are two units on the same site with separate control rooms, then an accident in one of the unit should not prevent the safe shutdown of the other unit. Transport mechanisms between the accident unit and the intakes to the operating unit control room should be reviewed for impact on CRH.

6.3 SMOKE INFILTRATION

The NRC Staff identified a concern that control rooms may be operated with significantly more inleakage than previously assumed and therefore control room operators may be

[2] Typically engineered safety feature (ESF) signals such as (1) containment high pressure or safety injection (SI), or (2) radiation monitors, or (3) both actuate control room isolation.

exposed to a greater amount of smoke infiltration in the event of a fire.. They hypothesized that, the increased smoke could make the control room uninhabitable and impair the operator's access to the remote shutdown locations. Although this is an extremely unlikely event, licensees should consider if they are appropriately prepared to mitigate such smoke infiltration.

Currently, no NRC regulations exist to establish smoke concentration limits or to define a design basis fire. The varied sources of fire compound the ability to identify the composition and amount of smoke. 10CFR50 Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979, does not provide guidance in this area. Although Issue 148, *Smoke Control and Manual Fire-Fighting Effectiveness*, of NUREG-0933, indicates that fires are anticipated to be of short duration.

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 ability to manage smoke infiltration into the control room. This guidance is provided in Appendix E. Performance of this recommended guidance should assure that the licensee could safely shut down the reactor in the event of smoke infiltration into the control room.

If the licensee qualitatively concludes that it does not have the ability to manage smoke, then it should take corrective actions to modify its smoke management capabilities.

6.4 TOXIC GAS EVALUATION

Control rooms are typically evaluated to assure that they can manage a toxic gas event. The last time this evaluation may have been performed was during the early 1980s in response to NUREG-0737 item III.D.3.4. One concern is that the amount of inleakage that the control room would experience during a toxic gas event may be greater than that assumed in the existing evaluation. This concern is addressed in Section 7. A second concern is that 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 Action

If it has been several years since the last toxic gas evaluation was performed, conduct an inventory of mobile and stationary sources of hazardous chemicals in the vicinity of the plant in accordance with Section 4.1 of Appendix H. If new toxic gas sources not previously evaluated are identified, revise the control room toxic gas evaluation in accordance with Appendix H.

7 AIR INLEAKAGE

7.1 PURPOSE

As discussed in Section 3.7, air in-leakage is one of numerous assumptions and/or design inputs used in the control room radiological and toxic gas evaluations for assessing Control Room Habitability. Over 20 percent of the control rooms were tested for in-leakage and each demonstrated that the actual measured value exceeded the value assumed in the accident analyses. This guidance 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 testing, testing, 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 I. The system assessment includes a walkdown to identify any discrepancies in the envelope and components vulnerable to in-leakage. The system assessment may be useful to identify potential in-leakage paths that are candidates for pre-test maintenance or modification. The licensee may choose to perform maintenance to eliminate any suspected in-leakage paths prior to performing a test for in-leakage.

7.3 TESTING

Licensees should perform a baseline test to determine the appropriate value for control room in-leakage for use in control room habitability analyses. The baseline in-leakage test may be performed using an integrated tracer gas test or performed using a component test methodology. A component test can be performed for a positive pressure control room by verifying that the control room pressure is greater than adjacent spaces. Additionally, a component test includes testing all components that cannot be verified to have a positive differential pressure relative to non CRE areas. Appendix J defines both methods further.

7.4 RESOLUTION OF IDENTIFIED ISSUES

Once a measured baseline in-leakage value is determined, it should be compared to the value used in the control room habitability radiological and toxic gas analyses. If the measured value is greater than the analyses input, the licensee should take corrective actions per its corrective action program (see Section 8). In addition, the measured in-leakage value should be evaluated from the perspective of smoke intrusion into the control room (see Appendix E). If a qualitative evaluation indicates a concern, actions should be taken to address the condition.

Corrective actions may include re-analysis, a design change and/or sealing and re-baseline testing to ensure regulatory requirements are met. Appendices C and D provides guidance that may be appropriate if re-analysis is desired. In addition, the alternative source term rule, 10 CFR 50.67, provides additional analysis methodology. Figure 3 reproduces the relevant portions of Figure 1 and demonstrates a process to follow.

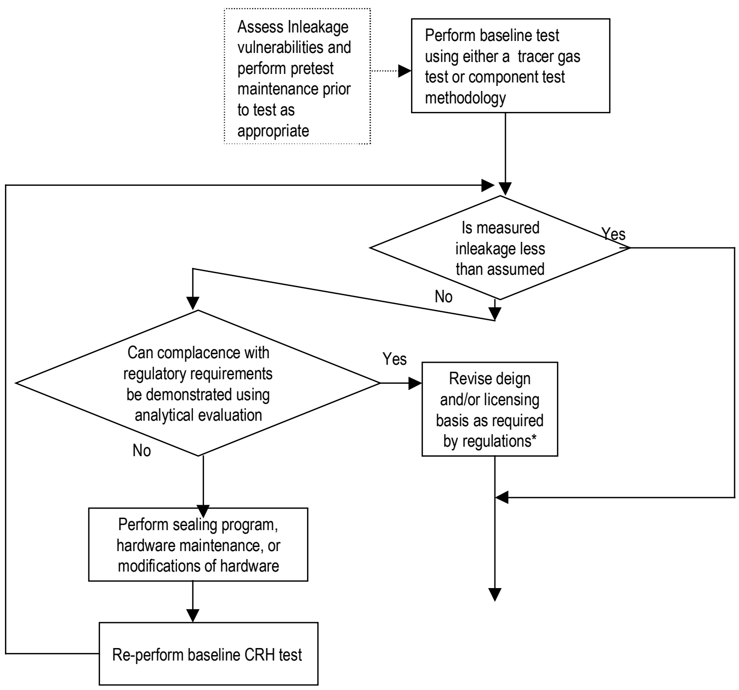

Figure 3
Inleakage Evaluation

8 METHODOLOGY FOR DISPOSITIONING AND MANAGING DISCREPANCIES

8.1 PURPOSE/SCOPE

Conditions adverse to safety or quality must be promptly identified and corrected in accordance with 10CFR50, Appendix B, Criterion XVI. This is accomplished by the 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

Generic Letter (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, but provides explicit insights on appropriate actions to take when a degraded or non-conforming 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 non-conforming conditions:

- The licensee may restore the structure, system, or component (SSC) to that which is described in the licensing basis. For example, if the assumed control room inleakage is explicitly described in the SAR and an inleakage test revealed excessive inleakage, the licensee may take corrective action to repair various seals and openings to reduce the inleakage to within the SAR limits.

- The licensee may accept a condition "as-is" which results in something different from that described in the SAR or may modify the plant to something different than that described in the SAR. 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 inleakage.
- The licensee may implement interim compensatory measures until permanent corrective actions can be fully implemented. These measures may be subject to 10CFR50.59. For instance, potassium iodide (KI) tablets and/or self contained breathing apparatus (SCBA) may be utilized to minimize operator dose until other actions are taken.

8.3 DETERMINING OPERABILITY AND REPORTABILITY

If a degraded or non-conforming condition is identified, appropriate action must be taken to maintain the plant in a safe condition. Technical Specifications establish the limits to assure safe operating conditions.

GL 91-18 provides detailed guidance with respect to performing operability determinations. As pointed out in Appendix J, it is advisable that contingency plans regarding operability determination and the justification for continued operating be completed before inleakage tests are performed. Such planning can provide insights about the baseline testing acceptance criteria. A licensee may want to determine:

- The level that could be accommodated within the current regulatory limits,
- The level that could be accommodated using the current source term, but with the:
  analysis improvements of Appendix C,
  the atmospheric dispersion improvements of Appendix D, and/or
  the compensatory measures of Appendix F (or other, plant-specific compensatory measures)
- The level that could be accommodated using the alternate source term (10CFR50.67 and Regulatory Guide 1.183) , but with the:
  - the atmospheric dispersion improvements of Appendix D, and/or
  - the compensatory measures of Appendix F (or other, plant-specific compensatory measures)

The reportability evaluation ensures timely reporting to the NRC of conditions or events significant 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 analysis, 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 which 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. As examples, 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 use of self-contained breathing apparatus and/or the availability of 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

Compensatory measures are limited to the short term. 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 Safety Analysis Report and for meeting the requirements of General Design Criterion 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 want to 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. Another option is to use the alternative radiological source term approach (i.e., based on 10 CFR 50.67 and Regulatory Guide 1.183).

An increase in previously calculated operator dose consequences of control room habitability due to reanalysis may require NRC review and approval. NEI 96-07, *Guidelines for 10CFR50.59 Safety Evaluations*, provides detailed industry guidance to address criteria for making this determination.

8.4.3 Repairing or modifying the plant

The identified inleakage source may be corrected by a repair of the physical discrepancy or by an improved sealing effort.

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

- Moving HVAC equipment within the boundary
- 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 were leakage sources when operating and pressurized

Modifications will most likely require a retest to ensure that the modification was successful in eliminating the inleakage condition. Re-testing may be conducted to verify the repair corrected the discrepancy and provide appropriate validation of the inleakage assumption.

PART 3 - ESTABLISHING AND MAINTAINING CRE INTEGRITY

9 LONG-TERM CONTROL ROOM HABITABILITY PROGRAM

9.1. Purpose

This section defines a program for maintaining control room envelope (CRE) integrity for control room habitability (CRH) over the life of the plant.

9.2. Control Room Integrity Program

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

Attributes of a CRE integrity program include:

- Periodic evaluations
  - System material condition
  - In-leakage challenges
  - Toxic gas challenges
- Configuration control (design and operation)
  - CRE barrier control
  - Procedure control
  - Design change control
  - Analyses change control
- Training

9.3. Periodic Evaluation

The baseline assessment and/or test program (per Sections 5, 6, and 7) demonstrated that the CR design, configuration, and operation met the design and licensing bases. However, control room envelope integrity is subject to change over time. A programmatic periodic evaluation of the systems, components, and key analysis assumptions is recommended to ensure optimal operation and early detection of problems. The periodic evaluation may involve design reviews, physical assessments, inspections, and/or testing as described below.

9.3.1. MATERIAL CONDITION

Testing of some of the mechanical CR HVAC components, such as emergency supply fans and charcoal filters, is normally required by Technical Specifications (see Section 9.5). However, degradation of various other components whose impact on CRH may be less obvious (e.g., door seals, manual dampers, control loops, non-safety related ventilation in adjacent spaces) can ~~significantly~~ affect safety analysis assumptions of CRE integrity. Also, passive components such as penetration seals and wall/floor joints can degrade over time.

The condition of these materials should be periodically assessed to ensure there is no degradation that would jeopardize integrity assumptions. In addition, for plants that assume pressurized CR design response in the analyses, the performance of HVAC systems in areas adjacent to the CRE should be reviewed to ensure that these spaces remain at a lower pressure than the CR. An inspection and testing program should be developed per Appendix K. An assessment frequency should be established based on material performance and service history.

9.3.2. In-Leakage Assessment

The design configuration and material condition of the CRE must be periodically assessed to ensure that CRE in-leakage assumption remains valid. This assessment may result in a need to retest CRE in-leakage. The frequency of the periodic assessment of the CRE is dependent on plant specific considerations as discussed below.

9.3.2.1. Assessment Frequency

The following factors should be considered when establishing the frequency for periodic assessment. This is not an inclusive list. Each plant must assess its own situation.

- Confirm that a control room envelope breaching program is in place. (Appendix L to NEI 99-03). Without a breaching program in place there can be no assurance that the boundary integrity will remain intact due to maintenance and/or modification activities.
- Determine the number of vulnerabilities. If the number of vulnerabilities is large then a more frequent assessment may be needed to show that these vulnerabilities are not impacting the in-leakage into the CRE.
- Determine the available differential pressure margin between the CRE and adjacent spaces. The typical differential pressure maintained by pressurized control rooms is 1/8 in wg. If the plant's measured differential pressure is smaller, more frequent assessments may be necessary to ensure that differential pressure is maintained.

- Compare the baseline test measured in-leakage to the design in-leakage assumption for both radiation and toxic gas considerations. A small margin may require a more frequent assessment.
- Confirm that maintenance practices are in place to assure that the boundary is maintained. This includes periodic inspections and preventative maintenance. (Appendix K of NEI 99-03).
- Confirm that the plant configuration control program evaluates the effect of modifications on CRE integrity.

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 in 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.

9.3.2.2. Determine Need to Test

If the CRE periodic assessment indicates that the actual 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. For example, plants with a large number of vulnerabilities and / or a small in-leakage margin may periodically perform testing to ensure that the actual in-leakage has not increased.

9.3.3. Toxic Gas Evaluation

Each plant typically performed a detailed evaluation of toxic gas sources in the vicinity of the plant during their initial licensing process. Many plants 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 outside of licensee control. It is recommended that each licensee should periodically perform a review of these sources in accordance with Section 4.1 of Appendix H.

Each licensee should establish a frequency for these periodic assessments, based on the number and type of industrial and transportation activities in the vicinity of the plant. 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. It is suggested that plants in industrialized areas perform such an assessment every 5 years; other plants 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 habitability. Plants are encouraged to review their existing programs and consider establishing new programs, if appropriate, to address the control aspects discussed below.

9.4.1 CRE Boundary / Barrier Control

Each plant should have a CRE Boundary Control program. Appendix L contains the guidance for establishing such a program if one does not already exist. The program assures 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 requires that each licensee establish a procedure control program. It is recommended that each plant review their 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 their chemical control program includes a review of new chemicals brought on site and that this review considers the potential impact of a release on the control room operator. It is recommended the program 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 requires that each licensee establish a design control program. This program ensures that the design bases are appropriately incorporated into the design and operation of the plant. This program also ensures that design changes, which includes permanent and temporary modifications, are subject to similar controls. Each plant should understand the design of the CRE and habitability systems and ensure that their program controls changes to the design. In addition, post modification testing as appropriate 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 such modification package.

9.4.5 Safety Analyses Control

Safety analyses are typically covered under a plant's design control program. Changes in the inputs and assumptions to these analyses that may or may not be related to a physical change, can affect the integrity of the CR. The analysis engineer should establish good communications with the system engineer to ensure system features are appropriately modeled.

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 and H.

9.5 TRAINING

The complexity and breadth of CRH warrants training of personnel. It is recommended that operations, maintenance, and engineering support personnel understand the bases for the CRE integrity program and issues that influence habitability. A satisfactory training program should ensure that those individuals associated with any aspects of this document recognize their role and its relationship to maintaining a habitable environment for the control room operator.

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10 REFERENCES

TO BE DEVELOPED.

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 (AEC) 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 for the facility 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*.

Appendix A was not retroactively applied to then-existing construction permit holders because § 50.34 is only applicable to *applicants* for construction permits.

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 during this period may have referenced it in their PSARs.

While the GDCs were under development, applicants proposed and the staff approved, various numeric criteria for the control room. As an example, at one plant the NRC approved the criterion of l0% of the Part 100 dose guidelines.

In the early 1970's, Murphy and Campe presented a method for evaluating radiological events in the control room. Additional information can be found in Reference 7 listed in Section 10 of this guidance document. 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.

2 THREE MILE ISLAND ACCIDENT EFFECT ON CONTROL ROOM HABITABILITY REQUIREMENTS

In 1979, following the Three Mile Island (TMI) accident, CRH was identified as a concern and NUREG-0660, *NRC Action Plan,* Item III.D.3.4, *Control Room Habitability Requirements,* was issued to document actions concerning the issue. In October 1980, the NRC issued NUREG-0737, which clarified the earlier requirements. In May 1982, the NRC issued Generic Letter 82-10, which requested licensees to confirm items completed and to propose specific schedule commitments for those items not yet completed, including Item III.D.3.4. In March 1983, the Commission issued orders to all power reactor licensees requiring them to confirm schedule commitments made in the licensees' responses to Generic Letter 82-10. The order included the following statement:

> *In accordance with Task Action Plan item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).*

Two classes of licensees were identified.

- Licensees with control rooms that meet the guidance of the Standard Review Plan (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.

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 (DBA LOCA). 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 deficiencies were identified in Licensee submittals, corrective actions were often deferred pending future resolution of certain issues such as the alternative source term (10CFR50.67).

3 REVIEWS OF CONTROL ROOM HABITABILITY IN THE 1980S

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

- ITEM B-66, Control Room Infiltration Measurements

- ISSUE 83, Control Room Habitability

The ACRS issued a letter to the Commission, on August 18, 1982, which:

- identified deficiencies in the maintenance and testing of engineered safety features designed to maintain control room habitability;
- provided examples of design and installation errors, including inadvertent degradation of control room leak tightness; and
- cited a shortage of NRC and licensee personnel knowledgeable about HVAC systems and nuclear air-cleaning technology. 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.

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. 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 Work Group was expected to identify any recommended actions that would correct significant deficiencies in control room habitability design, installation, test, or maintenance.

In June 1984, NRR provided a report to the EDO along with plans for implementing the recommendations of the report, including a survey of several operating plants. Based on the ongoing staff work, it was concluded that a solution had been identified and a schedule for the resolution of the issue was developed. NUREG/CR-4960 reported on the results of a 1986-1988 survey of control room habitability systems at twelve commercial nuclear generating stations. This survey was performed as part of the program discussed in the preceding paragraph intended to address concerns of the ACRS. The major conclusion of the report was that the numerous types of potentially significant discrepancies found among the surveyed plants could be indicative of similar discrepancies throughout the industry. In response to this report, the staff prepared a draft generic letter that called attention to the results of the survey and required licensees to perform and submit an assessment of their control room habitability. A draft letter was prepared but the effort was terminated during office concurrence.

As a result of the NUREG/CR-4960 studies, 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 and the HABIT Code was developed to provide an integrated code package for evaluating control room habitability. NUREG-1465, published in February 1995, provided updated source term information for the evaluation of control room designs.

As recommended by the ACRS, the staff was expected to consider NIOSH recommendations for toxic chemicals in its revision of Regulatory Guide 1.78.

4 TRACER GAS TESTING AND THE EVOLUTION OF AN INDUSTRY INITIATIVE

In recent years, several plants have conducted tracer gas testing to determine the amount of unfiltered inleakage into the control room envelope. The NRC reported the testing results at a July 16, 1998, public meeting on control room habitability. The testing data indicated that actual inleakage was much greater than the amount assumed in control room habitability analyses. Plants 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 inleakage 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 leak paths and represented documented measured inleakage rates. The constant injection tracer technique was considered the most useful method
- Well-managed sealing efforts are instrumental for assuring control room integrity
- Proper air flow balancing is essential to obtaining control room envelope and adjacent area HVAC system design basis

Following the joint public meeting with NEI, utility representatives, and representatives from the Nuclear HVAC Users Group (NHUG) in July 1998, the NRC staff was directed to work with the industry to resolve issues regarding control room habitability.

The Nuclear Energy Institute (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 August 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 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

2. REGULATORY GUIDES

The control room is expected to be habitable following design basis events. The design basis events that establish the bounding 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

  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 was 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 they provided to this issue, determine whether it still reflects the current design of the CRH features, and confirm that there is an safety evaluation report (SER) closing out the issue for their plants.

For a few plants, the NRC issued SERs that did not completely closeout TMI Action Item III.D.3.4 of NUREG-0737. Licensing documentation should be reviewed to ensure that no outstanding requests for additional information (RAIs) or SER issues exist.
The NRC permitted some plants with open TMI action items to use compensatory measures, such as the use of self-contained breathing apparatus or potassium iodide (KI) 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. This may involve additional analytical, design, or procedural change actions by these licensees to demonstrate compliance.

- 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 was much more thorough in outlining the requirements and acceptance criteria for each topic and also incorporated new regulatory positions, including several derived since the Three Mile Island incident (see NUREG-0737, discussed above).

  The SRP follows much the same outline as that for the Final Safety Analysis Report (at least those 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. As an example, an excerpt from NUREG-0800, Section 6.4:

> The LOCA source terms determined from the AEB 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.

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

  The US Atomic Energy Commission published TID-14844 in 1962 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. 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.

  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 possible relief for the issue of higher control room inleakage. 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 plant's licensing basis. These opportunities for relief will be described in greater detail in Appendix C.

- NUREG/CR-4960 – Control Room Habitability Survey of Licensed Commercial Nuclear Power Generating Stations

  NUREG/CR-4960 presents the results of a survey of twelve plants regarding the design of their systems used for control room habitability. The survey 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-5009 - (Extended Burn Fuel) (?????)

  NUREG/CR-5009 presents an assessment of the use of extended burnup fuel (to 60 GWd/t) in light water power reactors. The assessment was conducted by the Pacific Northwest Laboratory and published in February 1988. Accidents that involve the damage or melting of the fuel in the reactor core and spent-fuel handling accidents were reviewed. The report determined that the high burnup fuel rod gap Iodine-131 inventory may be 20 percent greater than the 10 percent value assumed in Regulatory Guide 1.25.

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

4. Inspection and Enforcement Notices

The following notices provided 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 86-76 – Problems Noted in Control Room Emergency Ventilation Systems
- IEN 92-18 – Potential for Loss of Remote Shutdown Capability during a Control Room Fire
- IEN 97-01 – Improper Electrical Grounding Results in Simultaneous Fires in the Control Room and the Safe Shutdown Equipment Room
- IEN 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 Voltage-based Repair Criteria
- IEN 99-05 - Inadvertent Discharge of Carbon Dioxide Fire Protection System and Gas Migration

APPENDIX C

CRH DOSE ANALYSIS: REGULATORY ENHANCEMENTS

Appendix C Table of Contents

CRH DOSE ANALYSIS: REGULATORY ENHANCEMENTS

1. PURPOSE AND SCOPE

This appendix provides guidance for performing control room dose calculations in support of control room habitability analyses for design basis radiological accidents. A fundamental commitment required for application of the NEI 99-03 methodology is to perform 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 offsite radiological releases. The DBA LOCA is generally the large break (LB) 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 for these events are required to identify the limiting event for the GDC 19 control room dose design criterion.

The events to be examined are those in the FSAR or licensee commitments. The assumptions used in the analyses of these events remain as stated in the plant licensing basis. For example, assumptions in the licensing basis analyses regarding concurrent loss of offsite power for accidents do not change.

2. GENERAL DESCRIPTION

Several areas of analysis have been examined to determine the appropriate methodology enhancements for control room dose analysis with application of the TID-14844 source term. This section's guidance may be used when applying the overall guidance contained in the remainder of NEI 99-03. The most important features are outlined in this section.

2.1. SOURCE TERM-SPECIFIC ISSUES

2.1.1. Gap release fractions for reactor transients with fuel damage and for fuel handling accidents were re-examined in light of the Alternative Source Term (AST) guidance development. The conclusion is that Table 3 of Regulatory Guide 1.183, Non-LOCA Fractions of Fission Product Inventory in Gap, with the additional considerations given in the accompanying Note 11, should be assumed for evaluation of non-LOCA accidents when used in conjunction with the maximum core radial peaking factor.

2.1.2. For fuel handling accidents, conservatism in assessing the number of failed pins should be consistent with levels of conservatism in other aspects of the analysis. Licensees may elect to submit a conservative analyses of the number of rods assumed damaged both for the spent fuel storage area and inside containment as described in SRP 15.7.4.

2.2. ISSUES RELATED TO IN-PLANT TRANSPORT AND RELEASE TO ENVIRONMENT

2.2.1. Containment leakage should be consistent with the expected containment pressure. Existing NRC regulatory guidance permits the reduction of assumed PWR containment leakage by a factor of two at 24 hours, and Regulatory Guide 1.183 extends this assumption to BWRs (including MSIV leakage) based on a reduction in the containment pressure over that period of time. Additional reductions may be justified on a plant-specific basis for the purpose of evaluating CRH.

Containment mixing during spray operation may take into account real phenomena like free convective mixing (i.e., density differences between the sprayed and unsprayed regions of containment) and momentum transfer between spray droplets and the containment atmosphere.

Suppression pool scrubbing credit may be taken per Standard Review Plan Section 6.5.5. Suppression pool scrubbing credit must account for the potential for suppression pool bypass and must use conservative (but not excessively conservative) flowrates from the drywell to the wetwell during core degradation. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool.

Credit for activity removal in secondary containment bypass pathways (including main steam lines) may be taken. As a minimum, the length of piping necessary to effect the bypass may be credited. Greater lengths may be credited on a case-by-case basis as long as the credited lengths are seismically qualified.

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 in whatever other programmatic document controls that allowable leakage.

For fuel handling accidents, even those involving high burn-up fuel, substantial pool decontamination factors (DFs) for radioiodine and other non-noble gas fission products are expected and should be credited. Pool pH and temperature, the concentration and the chemical form of the iodine released, and the presence of surrounding structures, with or without Safety-Related ventilation, may be considered when determining the fuel pool DF. A minimum fuel pool DF of 200 may be claimed for all iodine activity released if the depth of the water above the damaged fuel is 23 feet or greater. This value of 200 is an effective DF and is based on the assumption that no organic iodine decontamination occurs at all. The DF for fuel pools with less than 23 feet above the damaged fuel will be determined on a case-by-case basis.

3. COMMON FEATURES OF THE DOSE ANALYSIS

3.1. ‘GENERAL OUTLINE OF CONTROL ROOM HABITABILITY ANALYSIS

Description

During an accidental release of radioactivity, toxic gases, or smoke, contaminated air may enter the control room through inflow or inleakage into the control room ventilation system. The control room must remain habitable per 10CFR50 Appendix A GDC 19.

Regulatory Guidance

- USNRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800, "Control Room Habitability System", SRP 6.4 Rev. 2, July 1981.
- "Laboratory Testing of Nuclear-Grade Activated Charcoal", Generic Letter 99-02.

Deviations from Guidance and Clarifications

- The efficiencies of control room filtration systems should be those listed in the Technical Specifications. A safety factor of two must be included per GL-99-02.

- Automatic filtration initiation delay time should be based on Safety Injection Actuation System (SIAS) activation or radiation monitor response time, emergency diesel generator startup time, and a margin term.
- The control room dose limits are defined per 10CFR50 Appendix A GDC 19 and per SRP 6.4. A thyroid dose limit of 50 Rem is appropriate for use when the guidance in NEI 99-03 is followed. (See Section 5 of this Appendix)
- Upon a failure to meet regulatory requirements, a Justification for Continued Operation (JCO) relying on potassium iodide (KI) tablets and self contained breathing apparatus (SCBAs) may be utilized. The criteria to utilize KI tablets and SCBAs in JCO determinations may be incorporated in the FSAR or Technical Specifications. (See Appendix F)

3.2. COMMON ANALYSIS INPUT FOR CONTROL ROOM HABITABILITY ANALYSIS

Control room habitability dose analysis is based on a summation of doses resulting from primary containment leakage, Main Steam Isolation Valve (MSIV) leakage, Main Steam Safety Valve (MSSV) leakage, Atmospheric Dump Valve (ADV) leakage, Engineered Safety Feature (ESF) leakage, and direct shine from outside the control room envelope.

- The control envelope volume should be that as described in the plant's design basis.
- The control room filtered inflow should be taken from the design basis. When using design ventilation flow rates, the measurement uncertainty values that result in the highest calculated dose should be applied.
- The control room time-dependent unfiltered inleakage, considering appropriate uncertainties, should be determined. Prior to control room isolation, maximum (or nominal plus a margin) inflow should be used. A conservative time to isolation should be determined via appropriate testing or by assessing plant operating procedures. After control room isolation, inleakage should be determined based on appropriate testing applicable plant operating procedures.
- The relevant filter efficiencies used in the analysis should be per technical specifications.
- Standard working breathing rates of 3.5E-4 m^3/s and occupancy factors per Murphy-Campe and Standard Review Plan 6.4 respectively should be used. Justification for other values will be considered on an individual case basis.
- Control room recirculation and filtration may be credited. Recirculation flow should be at the technical specification minimum or, in the absence of technical specifications, as allowed by operating procedures.
- The control room operator dose due to direct shine from sources external to the control room are to be included in the total dose evaluation.

3.3. Core Power Level Assumptions

- Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 to account for power measurement uncertainties per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, 6/1/2000).

  At the time of the accident, 25% of all the equilibrium iodine fission products and 100% of the noble gas fission products are assumed available for release from the containment. This activity should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The iodine released to the containment is assumed to be composed of 91% elemental, 4% organic, and 5% particulate.

3.4. Accident Duration

Offsite doses should be determined over the accident duration and mitigation period described in the plant's UFSAR. Control room doses are evaluated over a 30-day period.

3.5. Miscellaneous Input parameters, Assumptions, and Initial Conditions

- 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 offsite power is assumed at the time of the accident.
- Dose conversion factors used in the analyses may be retained consistent with those described in the plant's licensing basis. Whole body doses have traditionally been based on semi-infinite cloud models, and thyroid doses have been based on DCFs presented in TID-14844, which are based on ICRP-2. Alternatively, as detailed in Section 5, the NRC staff agrees that thyroid dose conversion factors based on ICRP-30 such as those tabulated in Federal Guidance Report 11 are an acceptable change in methodology for application of NEI 99-03 guidance.
- 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.
- The atmospheric dispersion factors (χ/Qs) used in the analysis are derived for each release point. Credit for an elevated release can be calculated if the release height is more than two and one-half times the meets the elevated release criteria. (See Appendix D)

3.6. CONSEQUENCE ANALYSIS

The radiological consequence analysis described here is based on the NRC's traditional methods for calculating the radiological consequences of design basis accidents, which are described in a series of regulatory guides and Standard Review Plan chapters. That guidance was developed to be consistent with the TID-14844 source term and the whole body and thyroid dose guidelines stated in 10 CFR 100.11.

Specifically, the applicable regulatory guides are:

- 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 1.25, "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 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors"

Proposed deviations from this guidance will be discussed where applicable. The use of Alternative Source Terms (AST) is described in NUREG-1465 and Regulatory Guide 1.183. AST is not specifically addressed here.

4. SPECIFIC EVENT ANALYSIS FEATURES

This section describes features of specific events that should be evaluated for control room operator dose. The format is selected to present the modifications in analysis approach that can be applied in the context of NEI 99-03 guidance. Following each event description, the applicable regulatory guidance is recited consistent with accepted application of Standard Review Plan and TID-148444 source term analysis.

<u>Deviations from guidance</u> and clarifications to this guidance for NEI 99-03 application are identified for each event. Generic guidance, deviations from guidance, and clarifications that have been presented in Sections 2 and 3 are not repeated for each separate event. More detailed descriptions of analysis guidance is provided in Appendix G.

4.1. DESIGN BASIS ACCIDENTS

The design basis accidents (DBAs) evaluated for radiological consequences for offsite doses generally correspond to large break Loss of Coolant Accident (LBLOCA) scenarios when TID source term parameters are used. For control room habitability evaluations a spectrum of events must be analyzed to determine the limiting event with respect to control room operator dose. This section first outlines key features of the guidance modifications for the DBA LBLOCA events. Improvements in guidance for other classes of events, including coolant activity release accidents, DNB accidents, and fuel handling accidents (FHAs), are then described.

Accident Description

The DBA is typically the accident that results in the maximum amount of fuel damage. The regulatory guidance for this event is outlined in Regulatory Guides 1.3 and 1.4. For most plants, this is the LBLOCA event. Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 defines a LOCA as that postulated accident which results from a loss of reactor coolant inventory at a rate that exceeds the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe in the reactor coolant system are included. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core unless the water is replenished. With regard to radiological consequences, a LB LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment.

The individual contributions to the radiological consequences from a postulated LBLOCA are treated separately and then summed to obtain a total dose. These dose contributions include containment leakage (including the contribution from containment purge valves during closure), post-LOCA leakage from engineered safeguards feature (ESF) systems outside containment, Main Steam Isolation Valve (MSIV) / Main Steam Safety Valve (MSSV) / Atmospheric Dump Valve (ADV) leakage, and shine from sources outside the control room envelope. Other plant specific dose contributions may exist and should be evaluated.

4.1.1. PWR LARGE BREAK LOSS OF COOLANT ACCIDENT

The hypothetical design basis loss-of-coolant accident (LOCA) outlined in Regulatory Guide 1.4 is one of the postulated accidents used to evaluate the adequacy of structures, systems and components of a facility with respect to the public health and safety. The individual contributions to the radiological consequences from the hypothetical LOCA are treated separately and then summed to obtain a total dose. These dose contributions consist of containment leakage, post-LOCA leakage from ESF systems outside containment, including main steam isolation valves (MSIVs) and main steam safety valves (MSSVs), and shine from sources outside the control room envelope. Other plant specific dose contributions and pathways may exist and should be evaluated.

Regulatory Guidance

- USNRC, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors", Regulatory Guide 1.4, Rev. 2, June 24, 1974.

  Deviations from Guidance or Clarifications: Control Room Atmospheric Dispersion as discussed in Appendix D.

- USNRC, "Power Levels of Nuclear Power Plants", Regulatory Guide 1.49, Revision 1, December 1973.

  Deviations from Guidance and Clarifications: Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, 6/1/2000).

4.1.2. BWR LARGE BREAK LOSS OF COOLANT ACCIDENT

The hypothetical design basis loss-of-coolant accident (LOCA) outlined in Regulatory Guide 1.3 is one of the postulated accidents used to evaluate the adequacy of structures, systems and components of a facility with respect to the public health and safety. The individual contributions to the radiological consequences from the hypothetical LOCA are treated separately and then summed to obtain a total dose. These dose contributions consist of containment leakage, post-LOCA leakage from ESF systems outside containment, main steam isolation valves (MSIVs) and atmospheric dump valves (ADVs), and shine from sources outside the control room envelope. Other plant specific dose contributions and pathways may exist and should be evaluated.

Regulatory Guidance

- USNRC, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors", Regulatory Guide 1.3, Rev. 2, June 24, 1974.

  Deviations from Guidance or Clarifications: Control Room Atmospheric Dispersion as discussed in Appendix D.

- USNRC, "Power Levels of Nuclear Power Plants", Regulatory Guide 1.49, Revision 1, December 1973.

Deviations from Guidance and Clarifications: Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, 6/1/2000).

4.2. COOLANT ACTIVITY RELEASE ACCIDENTS

The subset of coolant activity release accidents is generally taken to be the PWR/BWR Main Steam Line Break (MSLB), the Steam Generator Tube Rupture (SGTR), and the Small Line Break Outside Containment. Since the last of these accidents is usually not limiting with respect to offsite or control room doses, this accident will not be addressed.

4.2.1. PWR MAIN STEAM LINE BREAK

Accident Description

The PWR Main Steam Line Break (MSLB) accident is a pre-trip guillotine-type rupture of a main steam line outside containment between the steam generator and the Main Steam Isolation Valve (MSIV). The increased rate of heat extraction by the affected steam generator causes a cooldown and depressurization of the reactor coolant system (RCS), which causes a positive reactivity addition with a negative MTC and FTC, causing core power level and heat flux to increase. Positive reactivity addition is terminated on CEA insertion post-SIAS. Turbine trip causes loss of AC (LOAC), which causes reactor coolant pumps (RCPs) to coast down, minimizing core flow, lowering DNBR and maximizing failed fuel pins. Cooldown of the RCS is terminated when the affected SG blows dry and AFW flow is isolated to the ruptured steam generator.

Regulatory Guidance

- "Steam System Piping Failures Inside and Outside of Containment (PWR)", SRP 15.1.5, Rev. 2, July 1981
- "Power Levels of Nuclear Power Plants", Regulatory Guide 1.49 Rev. 1, December 1973
- "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 1.25, March 23, 1993.
- “Calculation of Distance Factors for Power and Test Reactor Sites”, TID-14844, March 23, 1993.

<u>Deviations from Guidance and Clarifications</u>

- Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, June 1, 2000).
- The power peaking factor is per the COLR/Technical Specifications and not the factor of 1.65 per Regulatory Guide 1.25.
- The failed fuel fraction is that fraction of the fuel rods whose minimum DNBR is below the design limit or that fraction of the fuel rods, which exceed the minimum enthalpy limit per the current staff approach in NEI-99-03.
- The gas gap fractions are per Regulatory Guide 1.183: 8% I-131, 10% Kr-85, 5% other noble gases, 5% other halogens, and 12% alkali metals. For peak rod exposures <54 GWD/MTU or <62 GWD/MTU with peak rod average <6.3 kW/ft
- The MSLB creates an iodine spike in the primary RCS. The I-131 DEQ concentration in the RCS is estimated using a spiking model which assumes that the iodine release rate from the fuel rods to the RCS increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value stated in the Technical Specifications and lasts for the duration of the accident. A spiking duration of 3 hours is allowable unless a shorter time is justified.
- An expansion model of the affected steam generator blowdown may be assumed.
- Alternate Repair Criteria may be employed. The primary-to secondary leak rate and the primary I-131 dose equivalent (DEQ) concentration may be determined from the flex methodology of DG-1074.

4.2.2. BWR MAIN STEAM LINE BREAK

<u>Accident Description</u>

The BWR main steam line break (MSLB) accident description postulates a main steam line ruptures outside containment, releasing primary coolant activity into the turbine building. Two representative conditions for the primary coolant activity concentration are evaluated: (1) a pre-accident iodine spike is assumed depicted the condition where a reactor transient occurs prior to the accident and (2) the maximum equilibrium concentration permitted for continued full power operation is assumed.

Regulatory Guidance (With Departures and Clarifications)

- "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," Regulatory Guide 1.5 (Safety Guide 5), USNRC, Rev. 3/10/71.
  (a) This guide focuses on evaluating offsite doses. See general guidance below on evaluating control room doses.
  (b) Dose conversion factors based on ICRP 30 may be used instead of ICRP 2.
  (c) Atmospheric dispersion factors may be calculated using Appendix D
- "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Section 6.4, "Control Room Habitability System," USNRC, Rev. 2, July 1981.
  (a) The thyroid dose limit is 50 Rem instead of 30 Rem.

General Guidance on Calculating Control Room Dose for MSLB

- The activity within the reactor coolant and the total amount of coolant released from the break may be determined using Regulatory Guide 1.5.
- The radiological consequences of an MSLB may be evaluated by assuming that the reactor coolant that is released from the break forms a steam cloud that migrates towards the control room at a rate of 1 meter per second.
- The effect of buoyancy on the steam cloud transport may be credited with appropriate modeling assumptions.
- The activity concentration in the cloud may be determined assuming either Gaussian or uniform distribution.
- Based on the velocity and size of the steam cloud, the length of time required for it to pass by the control room may be determined.
- For the duration that the steam cloud is passing by the control room, the activity within the cloud may be drawn into the control room via filtered and unfiltered pathways, depending on plant-specific control room HVAC system response.
- Based on the time-dependent activity inside the control room, the 30-day dose to the operator is calculated.

4.3. DNB ACCIDENTS

There is a subset of accidents for which the source term consists of release of the gap activity from the fuel rods that sustain departure from nucleate boiling (DNB) or breach the critical power ratio (CPR) limit and are thereby predicted to sustain cladding failure. This is referred to as "failed fuel" (versus "defective fuel which leaks and causes the coolant activity release events described in the previous section), and results in the instantaneous release of the activity in the fuel rod gap and plenum to the reactor coolant system (RCS). These accidents are generally taken to be the Rod Ejection Accident (REA), the BWR Control Rod Drop Accident (CRDA), and the Locked Rotor Accident (LRA). These events are discussed here.

4.3.1. PWR ROD EJECTION ACCIDENT (REA)

Accident Description

For the postulated control rod ejection accident, a mechanical failure of a control rod mechanism housing is assumed such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position. The resulting reactivity insertion rate may result in a power burst capable of rupturing fuel pins, melting fuel, and could breach the primary system. Two release paths to the environment are evaluated: 1) Transport from Containment, and 2) Transport from Secondary System. The power burst may cause fuel failure due to a fuel enthalpy increase above a threshold value. If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this accident are then bounded by the consequences projected for the loss-of-coolant accident (LOCA), main steam line break, and steam generator tube rupture.

Regulatory Guidance

- USNRC, "Assumptions used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors ", Regulatory Guide 1.77, Rev. 0, May 1974.

- USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Section 15.4.8 Spectrum of Rod Ejection Accidents, and its associated Appendix A, Radiological Consequences of a Control Rod Ejection Accident, Rev. 2, July 1981.

  Deviations from Guidance or Clarifications: Control Room Atmospheric Dispersion as discussed in Appendix D.

- USNRC, "Power Levels of Nuclear Power Plants", Regulatory Guide 1.49, Revision 1, December 1973.

  Deviations from Guidance and Clarifications: Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, 6/1/2000).

4.3.2. BWR Control Rod Drop Accident

Accident Description

The control rod drop accident (CRDA) is the result of a postulated event in which a highest worth control rod drops from the fully inserted or intermediate position in the core. The highest worth rod becomes decoupled from its drive mechanism. The mechanism is withdrawn but the decoupled control rod is assumed to be stuck in place. At a later moment, the control rod suddenly falls free and drops to the control rod drive position (fully withdrawn). This results in the removal of large negative reactivity from the core and results in a localized power excursion.

For large, loosely coupled cores, this would result in a highly peaked power distribution and subsequent operation of shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion. The integrity of the turbine and condenser is unaffected by the rod drop accident.

The termination of this excursion is accomplished by automatic safety features of inherent shutdown mechanisms. Therefore, no operator action during the excursion is required.

Regulatory Guidance

- USNRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800, Section 15.4, Rev. 2, July 1981.

  Deviations from Guidance or Clarifications: Control Room Atmospheric Dispersion as discussed in Appendix D.

- USNRC, "Power Levels of Nuclear Power Plants", Regulatory Guide 1.49, Revision 1, December 1973.

  Deviations from Guidance and Clarifications: Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, 6/1/2000).

4.3.3. PWR Reactor Coolant Pump Rotor Seizure and Shaft Break

Accident Description

The accident is initiated by a seizure of the rotor or the break of the shaft of a reactor coolant pump (in PWR), which causes flow through the affected loop to be rapidly reduced. Some reverse flow may be expected through the affected loop (NUREG-800, SRP Sections 15.3.3-15.3.4). Reactor and Turbine trips occur with an assumed loss of offsite power. All the remaining reactor coolant pumps stop and cool down is performed by the operator releasing steam to the environment using the natural circulation emergency operating procedure. This is the limiting case, because with offsite power available the remaining coolant pumps would continue to operate and steam would be dumped to the condenser.

In current accident analyses for this event fuel failure is generally assumed to occur at the onset of DNB, even though the rods may be in a film boiling condition for a very short period of time. In fact, it is unlikely that fuel failure will occur. More advanced analysis of this event could demonstrate non-limiting results for CRH evaluations. A more detailed accident description and sequence of events is provided in Appendix G.

Regulatory Guidance

- USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Section 15.4, Rev. 2, July 1981.

  Deviations from Guidance or Clarifications: Control Room Atmospheric Dispersion as discussed in Appendix D.

- USNRC, "Power Levels of Nuclear Power Plants", Regulatory Guide 1.49, Revision 1, December 1973.

  Deviations from Guidance and Clarifications: Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, 6/1/2000).

4.4. Fuel Handling Accident

Accident Description

The fuel handling accident is analyzed to evaluate the adequacy of system design features and plant procedures provided for the mitigation of the radiological consequences of accidents that involve damage to spent fuel. Such accidents include the dropping of a single fuel assembly and handling tool or of a heavy object onto other spent fuel assemblies. Such accidents may occur inside containment, along the fuel transfer canal, and in the fuel handling building.

Regulatory Guidance

- "Radiological Consequences of Fuel Handling Accidents", SRP 15.7.4, Rev. 1, July 1981
- "Power Levels of Nuclear Power Plants", Regulatory Guide 1.49 Rev. 1, December 1973
- "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 1.25, March 23, 1993.
- "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, March 23, 1993.

<u>Deviations from Guidance and Clarifications</u>

- Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, June 1, 2000).
- The power peaking factor is per the COLR/Technical Specifications and not the factor of 1.65 per Regulatory Guide 1.25.
- The gas gap fractions are per Regulatory Guide 1.183: 8% I-131, 10% Kr-85, 5% other noble gases, 5% other halogens, and 12% alkali metals. For peak rod exposures <54 GWD/MTU or <62 GWD/MTU with peak rod average <6.3 kW/ft
- For fuel handling accidents, even those involving high burn-up fuel, substantial pool decontamination factors (DFs) for radioiodine and other non-noble gas fission products are expected and should be credited. Pool pH and temperature, the concentration and the chemical form of the iodine released, and the presence of surrounding structures, with or without Safety-Related ventilation, may be considered when determining the fuel pool DF. A minimum fuel pool DF of 200 may be claimed for all iodine activity released if the depth of the water above the damaged fuel is 23 feet or greater. This value of 200 is an effective DF and is based on the assumption that no organic iodine decontamination occurs at all. The DF for fuel pools with less than 23 feet above the damaged fuel will be determined on a case-by-case basis.
- 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.

5. Dose Analysis Limits

5.1. Control Room Operator Exposure

- ICRP-30 dose conversion factors are acceptable.
- Occupancy factors determining the most-exposed operator should consider actual plant staffing plans and measured doses to individual operators, including doses resulting from movement to and from the Control Room.

5.2. Dose Limits to Operators Consistent with Protecting Public Health and Safety

- Five Rem whole body or the equivalent dose to any part of the body is the current dose limit for the most-exposed Control Room operator.
- TEDE is an appropriate dose measure if all potentially dose-significant radionuclides are included in the dose analysis.
- 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."

Standard Review Plan 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 man 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.

For control room whole-body dose estimates, it is common to adjust the semi-infinite cloud DCF to account for the finite volume of the control room. 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 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 dose DCFs separately.

- International Commission Radiological Protection (ICRP) Publication 30, Pergammon Press, 1982.

6. Future Work

- Containment spray removal lambdas should take into account real phenomena like condensation on hygroscopic aerosols.

- Mixing in secondary containments or other structures which can be assumed to remain intact post-accident should not be limited only to periods when negative pressures have been achieved or only when such mixing is achieved by mechanical means (i.e., by ventilation and exhaust systems). Mixing credit may be taken on a case-by-case basis for CRH assessment whenever it can be shown that the potential for “short-circuiting” (i.e., direct leakage to the environment without substantial mixing) is unlikely.

- An acceptable level of overall conservatism (e.g., a 95th percentile Control Room dose) should be established. All contributing factors should be examined to ensure that, at least approximately, that level of conservatism is being achieved but not greatly exceeded. This may be more important for Control Room analysis than for offsite dose analysis because of the greater number of steps involved in the Control Room dose analysis and, therefore, the greater potential for excessive conservatism.

- [The following assertion is being considered for removal from this section. NRC Staff has indicated that credit for the CsI form of iodine is associated only with the AST; and absent credit for CsI, it appears unlikely that substantially lower release fraction than the 10 percent from the SRP can be justified.] Partition coefficients for radioiodine should reflect iodine chemistry as understood in light of its principal form being that of a cesium salt. This recognition affects both the ultimate iodine DF inside containment and the partitioning of iodine from leaked reactor coolant outside containment (i.e., as an option to using the SRP's 10 percent iodine release assumption).

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 (Control Room Dose Analysis). 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 H (Toxic Gas Assessments).

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. An overview of the NRC efforts in this area is presented in the following sections.

Note that licensees also have the option to reuse 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 Murphy and Campe in a paper presented at the 13th AEC Air Cleaning Conference (Reference 4.1). 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 (DBAs). The Murphy-Campe methodology has been in use since the mid-1970s and is the standard for most modern plants. Submittals using the Murphy-Campe methodology would most likely be subject to few questions regarding its use.

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. Though these wind tunnel and field tests had resulted in usable information for specific situations, Murphy and Campe acknowledged that general applicability was not possible. However, in order 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 DESCRIPTION

The Murphy-Campe methodology uses the following algorithm to predict atmospheric dispersion when the release is from a point source (e.g., a single vent or containment penetration) and the release point and receptor (e.g., control room intake) have a difference in elevation of less than 30% of the containment height:

$$\chi/Q = [3u\pi\sigma_y\sigma_z]^{-1}$$

If activity is assumed to leak from many points on the surface of the containment (i.e., area source) in conjunction with a single point receptor (or the release is from a single point source and the release point and receptor have a distance in elevation greater than 30% of the containment height), the Murphy-Campe methodology is then based on the following algorithm:

$$\chi/Q = [(\pi\sigma_y\sigma_z + cA)u]^{-1}$$

where A is the projected area of the containment building and c is a building wake coefficient. The building wake coefficient c is a function of the distance s between the containment surface and the receptor location and d is the diameter of the containment as follows:

$$c = [2 + 3(d/s)^{1.4}]^{-1}$$

Note that c approaches zero as s/d becomes small and approaches ½ for large s/d. This expression was deduced from early wind tunnel data on a model of a rounded containment building. Using this expression on block-like structures is questionable because of the different flow characteristics associated with rounded structures.

2.1.3 IMPLEMENTATION

The Murphy-Campe methodology consists of first determining the five percentile χ/Q value (defined as the χ/Q value which is exceeded 5% 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 m/sec. The determination of χ/Q values for subsequent time intervals involves corrections for long term meteorological averaging for wind speed and wind direction.

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 NRC decided to consider the feasibility of identifying or developing a more robust methodology that would better describe atmospheric dispersion near buildings. The Pacific Northwest National Laboratory under contract with the NRC subsequently developed ARCON96 (Reference 4.2). The model was developed using detailed meteorological field data and, as such, showed improved performance in predicting the effect of building wakes, particularly under light wind conditions.

2.2.2 DESCRIPTION

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 releases 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 4.3) and PAVAN (Reference 4.4). 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 4.5). The experiments were conducted under a wind 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 five percent of the time are determined from the cumulative frequency distributions for each averaging period. Finally, the relative concentrations for five standard averaging periods used in control room habitability assessments are calculated from the five percentile relative concentrations.

Note that although ARCON96 is based on a simple Gaussian dispersion model, the χ/Q values predicted by the model do not very 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 m/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 m/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. Guidance for running ARCON96 follows.

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 are maintained under the site's meteorological measurements program as described in the facility's licensing basis. Five years of hourly observations should be used. If less 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 I 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 many situations, ground level releases can be better classified as area sources. Examples 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 receptor (i.e., control room intake) and treated 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 receptor (i.e., control room intake). In the absence of site-specific empirical data, the initial diffusion coefficients are found by:

  σ_y = (area source width)/6

  σz = (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

  $\sigma_z = 0.0$

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; otherwise, the release should be considered a ground level release. Use of the elevated plume option may lead to unrealistically low concentrations at control room intakes located 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.[3]

If addressed in the current licensing basis, 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 (Reference 4.6).[4] Ground level χ/Q values generated by ARCON96 may be substituted for values generated with Equation 5 of Regulatory Guide 1.145.

2 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 must encompass a sufficient range of meteorological 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.

3 WIND TUNNEL TESTS

Wind Tunnel testing is a widely used and accepted approach in a number of industries. As mentioned previously, the Murphy-Campe methodology is based on a number of wind tunnel and field tests. Wind tunnel test results have also been used to benchmark ARCON96. The use of site-specific wind tunnel test results could be an alternate approach to developing atmospheric dispersion factors (χ/Qs). The acceptance of such an approach would be linked to an effort that would demonstrate the applicability of the existing database of test results for the non-nuclear industries to a nuclear power plant. Utilizing measured maximum values of the atmospheric dispersion, those individual plants that have performed the testing to date have seen more realistic χ/Qs and a significant reduction in calculated control room doses following an event. The cost of this testing can also be significantly less than other approaches. A joint industry initiative, similar to the development of this document, specifically for the use of wind tunnel testing, would allow the appropriate regulatory approval and subsequent application of this approach on a site-specific basis.

4 REFERENCES

4.1. NUREC/CR-6331, Revision 1, Atmospheric Relative Concentrations in Building Wakes, U.S. Nuclear Regulatory Commission, Washington D.C.

[3] 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.

[4] 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.

4.2. NUREG/CR-2919, XOQDOQ: Computer Program for the Meteorological Evaluation of Routine Releases at Nuclear Power Stations, U.S. Nuclear Regulatory Commission, Washington D.C.

4.3. NUREG/CR-2858, PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations, U.S. Nuclear Regulatory Commission, Washington D.C.

4.4. PNL-10286, *Atmospheric Dispersion Estimates in the Vicinity of Buildings,* Pacific Northwest Laboratory, Richland, Washington.

4.5. CONF-740807, Murphy, K.G., and K.M. Campe. *Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19.* In *Proceedings of the 13th AEC Air Cleaning Conference,* August 12-15, 1974, San Francisco, California, U.S. Atomic Energy Commission, Washington, D.C.

4.6. Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, Revision 1, Reissued February 1983, U.S. Nuclear Regulatory Commission, Washington D.C.

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.
	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.
	Elevation Difference (meters)	This value is used to normalize the release heights and the receptor heights in those cases were 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 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.

Parameter Type	Parameter	Discussion	Acceptable Input
	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 time the height of structures in the vicinity of the stack, the release should be classified as ground.
	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.[5]

[5] 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.

Parameter Type	Parameter	Discussion	Acceptable Input
	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.
	Stack Flow (m^3/sec)	This value is used for all release models to ensure that the near field concentration are not grater 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 50 m^3/sec.
	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.

Parameter Type	Parameter	Discussion	Acceptable Input
	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.
	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 σ_z = 0.0 Otherwise, use σ_y=σ_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 SHUT DOWN

1. OBJECTIVE

The objective of this Appendix is to provide a qualitative assessment tool for managing the issue of smoke infiltration as described in Section 6 of this document. The goal is to ensure that the operator maintains the ability to safely shut down the plant during a fire/smoke event originating outside the control room.

2. ASSESSMENT

An assessment should be performed to determine any significant barriers that prevent achieving the stated goal. This assessment should determine if a single credible fire/smoke event has the potential to simultaneously prevent the operator from shutting down the plant from both the Control Room and the Remote Shutdown locations. The following items should be reviewed:

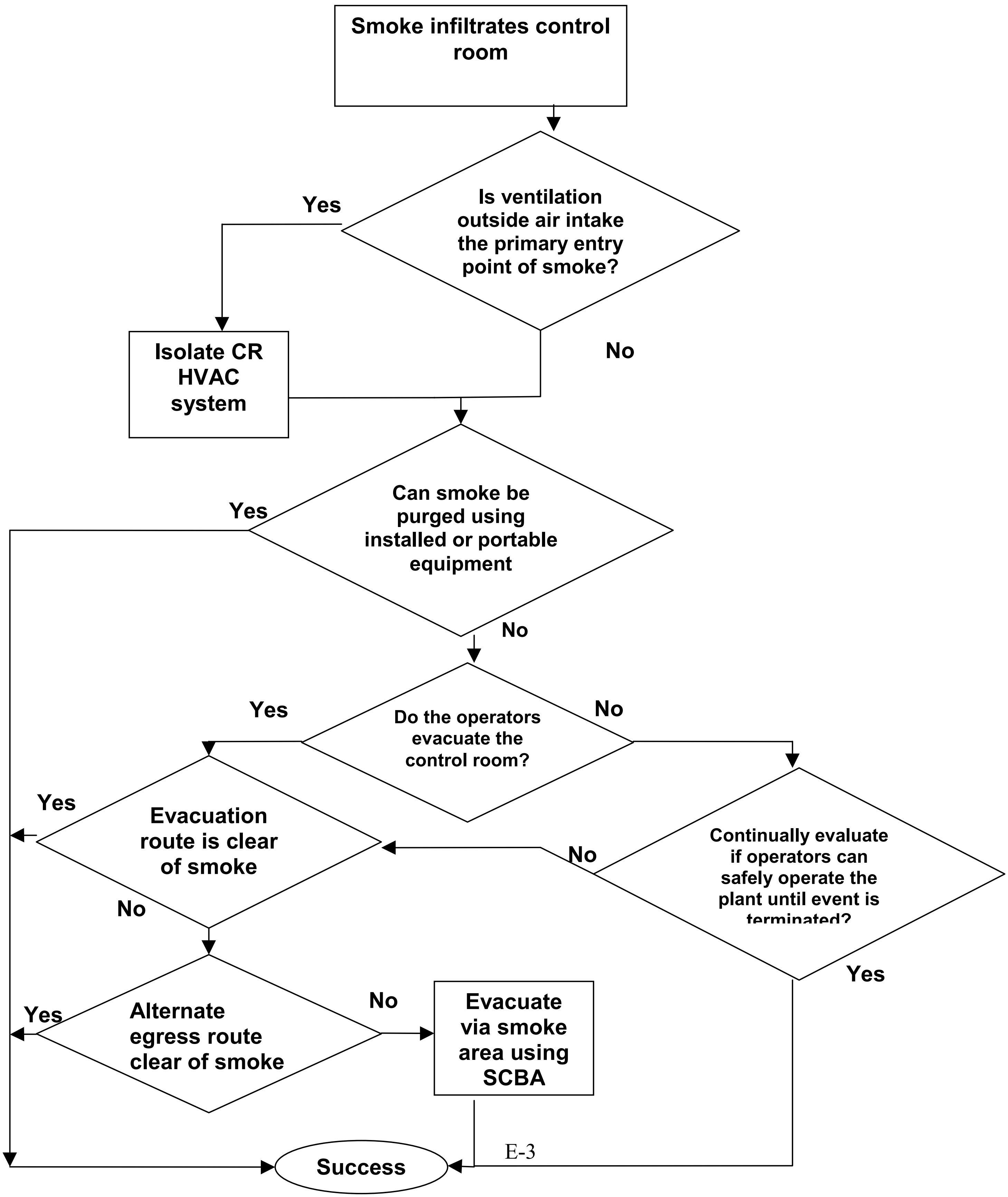
- Verify the Remote Shutdown Panels are not located within the Control Room Habitability envelope.
- Verify the Remote Shutdown Panels and the Control Room are adequately separated by distance, or appropriate fire barriers, such that a single credible fire/smoke event in one area could not affect the habitability of the other.
- Verify 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. If not, verify an alternate egress path exists and 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 draft smoke away from the Control Room when practical.
- Verify that a sufficient number of Control Room Operators per shift are qualified in the use SCBA's to safely shutdown the plant. Certain success paths to achieve the stated goal above may require the limited use of SCBAs.
- Verify that the appropriate SCBA and smoke removal equipment is 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 some 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. A flow chart of the process is contained in Figure E.1.

- 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 SCBA's 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. This may require the continued use of SCBA's. Consideration should be given to evacuate to the Remote Shutdown Panels. This decision would be based on the severity of the situation (smoke concentration) and the availability of a safe egress path to the remote shutdown panels.
- If the Remote Shutdown Panels 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 the primary or alternate path to the Remote Shutdown Panel that is least effected by the event.It may be necessary to use SCBA's while transiting to the Remote Shutdown Panel.
- 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.

Figure E.1


Smoke infiltrates control room
Is ventilation outside air intake the primary entry point of smoke?
Yes
No
Isolate CR HVAC system
Can smoke be purged using installed or portable equipment
Yes
No
Do the operators evacuate the control room?
Yes
No
Evacuation route is clear of smoke
Yes
No
Continually evaluate if operators can safely operate the plant until event is terminated?
No
Yes
Alternate egress route clear of smoke
Yes
No
Evacuate via smoke area using SCBA
Success

APPENDIX F

COMPENSATORY MEASURES ALLOWABLE ON AN INTERIM BASIS

1. 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. These are use of self-contained breathing apparatus and use of potassium iodide tables. Other plant specific compensatory actions may be appropriate. The use of any compensatory measure will required a plant specific evaluation to justify its use.

The use of SCBAs and KI has been determined to be acceptable for addressing CRE integrity in the interim situation until the licensee remediates the issue. However, use of SCBAs or KI in the mitigation of situations where inleakage does not meet design basis limits is not acceptable as a permanent solution. 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 must be developed.

2. USE OF SCBAs AS A COMPENSATORY MEASURE

Short-term credit for SCBA use to support control room habitability assessments should be allowed generically, provided an approved respiratory protection program is in effect. Application of long term credit will be reviewed on an individual case basis. However, prior to acceptance of this credit, licensees must address several points. Several of these points may require commitments to be made as indicated below. Each individual licensee must determine how these commitments are to be made and met.

2.1. KEY CONSIDERATIONS FOR CREDITING SCBA USE IN SUPPORT OF CONTROL ROOM HABITABILITY ASSESSMENTS

1. **An approved respiratory protection program must be in effect.**
   a) An approved respiratory protection program in accordance with Regulatory Guide 8.15 and NUREG-0041 is established and in place.
      i) Maintaining an adequate respiratory protection program is vital to their 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 IDLH conditions. Therefore, non-radiological hazards must 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

2. Sufficient number of operators must be trained and qualified in SCBA use.

a) The licensee must commit to 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 during SCBA use.

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

d) There must be effective program oversight or controls in place for tracking and maintaining operators' required periodic retraining and SCBA fit testing.

3. Adequate supplies of equipment must be available.

a) There must be sufficient numbers of dedicated, surveyed, and inventoried equipment with various size face pieces available for use by control room operators at all times.

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

4. Corrective lenses (if required) must be available for SCBA users.

a) The site must make a commitment that all qualified users will have the necessary corrective lenses (either approved mask spectacles or soft contact lenses) available in the control room while on duty.

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 contact lenses may be worn if their use has been satisfactorily demonstrated.

e) Hard contact lenses shall 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.

5. Persons using tight fitting (facepiece) respirators shall not have any facial hair that interferes with the sealing surfaces of the respirator.

a) The licensee must commit to have a minimum number of control room operators qualified in SCBA use to be clean-shaven while on duty.

b) Those operators on duty who are not clean shaven must either exit the control room or shave before the time that SCBA use would be required (or at the onset of a radiological emergency). Shaving should not be rushed, as this could cause open wounds.

c) Any intrusion of facial hair into the sealing surface results in an increase in leakage due to fit degradations, interference with proper operation of SCBA facepiece components, and a shortened period of air supply. This could lead to degraded operator emergency response.

6. Adequate method(s) to refill SCBA air bottles must 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 must be properly monitored and attended to ensure that the air intake remains in an uncontaminated atmosphere.

7. Provide for adequate relief from respirator use.

a) Provisions must be made for operators wearing SCBA to leave the area for relief in case of equipment malfunction, undue physical or psychological distress, procedural or communication failure, significant deterioration of operational conditions, or any other condition that might require such relief.

b) The result of wearing SCBA is to substantially reduce worker efficiency due to physical stress and the relatively short working time limited by air tank capacity.

c) The periods of time respirators are worn continuously and the overall duration of use should each be kept to a minimum.

i) Assignment of specific time limits on respirator use is difficult due to the wide variations in job requirements and in the physical capacities and psychological attitudes of individuals.

ii) Air may be used more rapidly than a rating indicates, particularly under the stress of an emergency. The duration of SCBA use will depend on:

a) The degree of physical activity of the user

b) The physical condition of the user

c) The degree to which the user's breathing is increased by excitement, fear, or other emotional factors

d) The degree of training or experience that the user has had with this or similar equipment

e) Whether or not the cylinder is fully charged at the start of the work period

f) The presence in the compressed air of carbon dioxide concentrations greater than the 0.4% normally found in atmospheric air

g) The condition of the apparatus

8. Ensure an appropriate monitoring program exists.

a) An appropriate air sampling program must 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 only to airborne radiological hazards and may not be appropriate to circumstances when chemical or other respiratory hazards exist instead of, or in addition to, radioactive hazards.

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.

a) In the case of fire fighters, exposure to radioactive materials is generally of secondary importance; toxic fumes/gases are the principal hazard.

b) However, a strict legal reading of the regulations leads us to conclude that nothing prohibits using post-work whole body counts for demonstrating compliance with Part 20 limits.

3. USE OF POTASSIUM IODIDE (KI) 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 potassium iodide (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 ten 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). In order 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:

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.
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. Personnel who are identified to be required 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.

4. 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.
5. Plant procedures should be in place to direct administration of KI to control room personnel within two hours of a radioiodine release. Procedures must also be in place to administer KI to on-coming shifts as necessary if radioiodine releases continue.
6. 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. NUREG-0737, Task III.D.3: Worker Radiation Protection Improvement (Rev. 3), TMI Action Item III.D.3.2 (4), "Develop Air Purifying Respirator Radioiodine Cartridge Testing and Certification Criteria"
2. 10 CFR 20 (RIN 3150-AF81), "Respiratory Protection and Controls to Restrict Internal Exposures"
3. 10 CFR Part 20 Appendix A - Assigned Protection Factors (APF) for Respirators
4. NRC Information Notice 98-20, "Problems With Emergency Preparedness Respiratory Protection Programs"
5. Regulatory Guide 8.15, "Acceptable Programs For Respiratory Protection"
6. NUREG-0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials"
7. NRC Information Notice 99-05: Inadvertent Discharge Of Carbon Dioxide Fire Protection System And Gas Migration (March 8, 1999)
8. HPPOS-094 (PDR-9111210195): Guidance Concerning 10 CFR 20.103 and Use of Pressure Demand SCBA's
9. HPPOS-116 (PDR-9111210272): OSHA Interpretation: Beards and Tight-Fitting Respirators
10. NUREG/CR-1433, 1980, "Examination of the Use of Potassium Iodide (KI) as an Emergency Protective Measure for Nuclear Reactor Accidents," David C. Aldrich and Roger M. Blond.
11. NUREG/CR-6310, 1995, "An Analysis of Potassium Iodide (KI) Prophylaxis for the General Public in the Event of a Nuclear Accident," H. Behling, K. Behling and H. Amarasooriya.

12. NCRP Report No. 55, August 1, 1977 “Protection of the Thyroid Gland in the Event of Releases of Radioiodine.”

APPENDIX G

CRH DOSE ANALYSIS: DETAILED GUIDANCE

Appendix G Table of Contents

CRH DOSE ANALYSIS: DETAILED GUIDANCE

1.0 PURPOSE AND SCOPE

This appendix is designed to provide additional guidance in the CRH dose analysis process. At this time the following provisions apply to this guidance:

- This appendix is a "work-in-progress". Both the NRC and the NEI task force believe that additional guidance regarding performance of CRH dose analyses would be very valuable to licensees. At the same time, it is recognized that a thorough prescription for these analyses is far beyond the scope of this current level of effort. This appendix merely compiles some analysis guidelines that have been compiled by the task force. It is neither complete nor comprehensive.
- This appendix will not be reviewed by the NRC and, therefore, will not become a part of any NRC endorsement of the NEI 99-03 document.
- Both the task force and the NRC agree that this type of guidance is valuable. The NRC staff has expressed interest in future efforts that would culminate in uniform, approved methodologies and assumption sets for CRH dose analysis, as well as for off-site dose analysis.
- The current structure provides an expansion of the event descriptions that are presented in Appendix C, Section 4. In this case the intent is to assure that all material is repeated here in some context. Any changes made in Appendix C, Section 4 should be incorporated here in the appropriate subsection(s).

2. SPECIFIC EVENT ANALYSIS FEATURES

2.1. DESIGN BASIS ACCIDENTS

The design basis accidents (DBAs) evaluated for radiological consequences for offsite doses generally correspond to large break Loss of Coolant Accident (LBLOCA) scenarios when TID source term parameters are used. For control room habitability evaluations a spectrum of events must be analyzed to determine the limiting event with respect to control room operator dose. This section first outlines key features of the guidance modifications for the DBA LBLOCA events. Improvements in guidance for other classes of events, including coolant activity release accidents, DNB accidents, and fuel handling accidents (FHAs), are then described.

<u>Accident Description:</u>

The DBA is typically the accident that results in the maximum amount of fuel damage. The regulatory guidance for this event is outlined in Regulatory Guides 1.3 and 1.4. For most plants, this is the LBLOCA event. Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 defines a LOCA as that postulated accident which results from a loss of reactor coolant inventory at a rate that exceeds the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe in the reactor coolant system are included. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core unless the water is replenished. With regard to radiological consequences, a LB LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment.

The individual contributions to the radiological consequences from a postulated LBLOCA are treated separately and then summed to obtain a total dose. These dose contributions include containment leakage (including the contribution from containment purge valves during closure), post-LOCA leakage from engineered safeguards feature (ESF) systems outside containment, Main Steam Isolation Valve (MSIV) / Main Steam Safety Valve (MSSV) / Atmospheric Dump Valve (ADV) leakage, and shine from sources outside the control room envelope. Other plant specific dose contributions may exist and should be evaluated.

2.1.1. PWR Large Break Loss of Coolant Accident

The hypothetical design basis loss-of-coolant accident (LOCA) outlined in Regulatory Guide 1.4 is one of the postulated accidents used to evaluate the adequacy of structures, systems and components of a facility with respect to the public health and safety. The individual contributions to the radiological consequences from the hypothetical LOCA are treated separately and then summed to obtain a total dose. These dose contributions consist of containment leakage, post-LOCA leakage from ESF systems outside containment, including main steam isolation valves (MSIVs) and main steam safety valves (MSSVs), and shine from sources outside the control room envelope. Other plant specific dose contributions and pathways may exist and should be evaluated.

Regulatory Guidance

- USNRC, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors", Regulatory Guide 1.4, Rev. 2, June 24, 1974.

  Deviations from Guidance or Clarifications: Dose Methodology as discussed under General Outline of Control Room Habitability Analysis in this Appendix C.
  Control Room Atmospheric Dispersion as discussed in Appendix D.

- USNRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800.

  Deviations from Guidance and Clarifications: It is not necessary to assume the gross failure of a passive component previously required in Appendix B of

SRP Section 15.6.5 for those plants that don't provide an ESF atmosphere filtration system.

- USNRC, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.
- USNRC, "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," Regulatory Guide 1.52, Revision 2, March 1978.
- "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, March 23, 1962.
- USNRC, "Laboratory Testing of Nuclear-Grade Activated Charcoal," NRC Generic Letter 99-02, June 3, 1999.
- USNRC, "Potential Radioactive Leakage to Tank Vented to Atmosphere," Information Notice 91-56, September 19, 1991.
- USNRC, "Power Levels of Nuclear Power Plants", Regulatory Guide 1.49, Revision 1, December 1973.
  Deviations from Guidance and Clarifications: Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, 6/1/2000).

2.1.2. BWR LARGE BREAK LOSS OF COOLANT ACCIDENT

Primary Containment Leakage Contribution Assumptions

This dose contribution is due to leakage from the primary containment to the atmosphere through various pathways. For BWRs, it is mostly into the secondary containment that is typically filtered via the Standby Gas Treatment System (SGTS). However, other potential pathways must also be considered and evaluated. The release during the drawdown period may also need to be evaluated without filtration. Accident mitigation systems (e.g., sprays, suppression pool scrubbing, and filtration) are designed to reduce the source term available for release to the environment. Leakage through MSIVs will be treated separately.

The primary containment leakage rate is assumed to remain constant over the course of the accident. However, the NEI 99-03 guidance specifies that this leakage will be reset to one half the initial leak rate after 24 hours for a BWR (similar to the assumptions for a PWR).

Primary containment leak rates of less than 0.1% per day typically have not been accepted by the NRC staff due to integrated containment leakage test sensitivity limitations. The leakage rate used in the analysis should correspond to that given in the technical specifications.

Noble gas releases to the environment are unaffected by the presence of filters or sprays.

Reduction in containment airborne radioactivity by containment spray systems that have been designed and maintained in accordance with SRP 6.5.2 may be credited. The mixing rate attributed to natural convection between sprayed and unsprayed regions, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed region per hour, unless other rates are justified. The containment atmosphere may be considered a well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown.

- 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.
- Reduction in airborne radioactivity by suppression pool scrubbing should be credited per SRP 6.5.5. If the time-integrated decontamination factor (DF) values claimed for removal of particulates and elemental iodine are 10 or less for a Mark II or a Mark III containment, or are 5 or less than for a Mark I containment, these values may be accepted. A DF of one (no retention) should be used for noble gases and organic iodines. Justification for greater DF values will be considered on an individual case basis.
- Where dilution credit for a secondary containment with recirculation is claimed, adequate mixing in the secondary containment volume should be demonstrated.
- Secondary containment bypass leakage must be evaluated. This leakage, usually expressed as a fraction or percentage of the primary containment leak rate, is assumed to pass from the primary containment directly to the environment, bypassing secondary containment. This leakage rate is specified in the UFSAR.

ESF Leakage Contribution Assumptions

This contribution includes postulated leakage from ESF components to include the leakage from valve stems and pump seals that can be expected during the operation of the ESF recirculation systems as well as the leakage from a postulated gross failure of an ESF passive failure such as the failure of a pump seal.

- The leakage used for calculating the radiological consequences should be the maximum operational leakage and should be taken as two times the sum of the simultaneous leakage from all components in the recirculation system above which the technical specifications would require declaring such systems to be out of service. This leakage is assumed to occur throughout the accident, starting at the earliest time that the recirculation mode is initiated and ending at the latest time the releases from these systems are terminated.

- The airborne iodine is assumed to be released immediately to the environment. ESF atmosphere filtration credit, where applicable, may be taken in those areas where such leakage is postulated to occur in order to mitigate the radiological consequences from the fission product release.
- 50% of the core iodine inventory should be assumed to be mixed in the sump water being circulated through the containment external piping systems.
- Credit may be taken for radioactive decay of the iodine during the time period from the occurrence of the LOCA up to the beginning of recirculation when the sump water is circulated outside the containment.
- For a sump water temperature above 212 F, the fraction of the leakage that flashes to steam is determined assuming a constant enthalpy process. If the flash fraction is greater than 10%, then this fraction is used. If the calculated flash fraction is less than 10% or if the water is less than 212 F, then 10% of the iodine in the leakage is assumed to become airborne unless a smaller amount is justified based on actual sump pH history and ventilation rates.

MSIV Leakage Contribution Assumptions

This contribution is treated similar to a primary containment bypass pathway. However, since the dose consequences are more significant, it is treated separately. Credit for non safety-related equipment should be applied carefully and will be reviewed on an individual case basis.

- 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, but not less than 50% of the maximum leak rate, after the first 24 hours if supported by site-specific analyses.
- 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. It is assumed to be instantaneously distributed in the drywell free volume at the time of the accident. No credit for leakage of activity from the drywell to the containment (Mark III) or to the suppression pool region (Mark I and II) is assumed. No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel. Credit may be assumed for radioactive decay of the fission products in the drywell prior to operation of the MSIVLCS.
- No release of activity from the MSIV Leakage Control System (MSIVLCS) is assumed up to the time of system actuation.

- Leakage through valve stems or drain lines to an untreated region is assumed to immediately be released to the atmosphere without holdup or mixing credit.
- MSIV releases that are directed to treated regions are assumed to be directly to the filter intake unless flow is mechanically directed to a distribution header. Credit for mixing is given on the same basis as for other leakage to this system.
- 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.
- Reduction in MSIV releases that are due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, 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 (SSE). The amount of reduction allowed will be on an individual case basis.
- For cases where the turbine and condensers can be credited, leakage to the atmosphere from the turbine and condensers is at a rate of 1% per day unless a larger value is shown based on input flow rates. Credit can be assumed for radiological decay during holdup in the turbine and condensers.
- In the absence of collection and treatment of releases by ESFs such as the turbine and condensers or a MSIVLCS, leakage should be assumed to be immediately released to the environment as an unprocessed, ground level release.
- Holdup and dilution in the turbine building should not be assumed.

Regulatory Guidance

- USNRC, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors", Regulatory Guide 1.3, Rev. 2, June 24, 1974. **[Revision # and date?]**

  Deviations from Guidance or Clarifications: Dose Methodology as discussed under General Outline of Control Room Habitability Analysis in this Appendix C.
  Control Room Atmospheric Dispersion as discussed in Appendix D.

- USNRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800.

  Deviations from Guidance and Clarifications: It is not necessary to assume the gross failure of a passive component previously required in

Appendix B of SRP Section 15.6.5 for those plants that don't provide an ESF atmosphere filtration system.

- USNRC, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.
- USNRC, "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," Regulatory Guide 1.52, Revision 2, March 1978.
- "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, March 23, 1962.
- USNRC, "Laboratory Testing of Nuclear-Grade Activated Charcoal," NRC Generic Letter 99-02, June 3, 1999.
- USNRC, "Potential Radioactive Leakage to Tank Vented to Atmosphere," Information Notice 91-56, September 19, 1991.
- USNRC, "Power Levels of Nuclear Power Plants", Regulatory Guide 1.49, Revision 1, December 1973.

Deviations from Guidance and Clarifications: Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, 6/1/2000).

2.2. Coolant Activity Release Accidents

The subset of coolant activity release accidents is generally taken to be the PWR/BWR Main Steam Line Break (MSLB), the Steam Generator Tube Rupture (SGTR), and the Small Line Break Outside Containment. Since the last of these accidents is usually not limiting with respect to offsite or control room doses, this accident will not be addressed.

2.2.1. PWR Main Steam Line Break

Accident Description

The PWR Main Steam Line Break (MSLB) accident is a pre-trip guillotine-type rupture of a main steam line outside containment between the steam generator and the Main Steam Isolation Valve (MSIV). The increased rate of heat extraction by the affected steam generator causes a cooldown and depressurization of the reactor coolant system (RCS), which causes a positive reactivity addition with a negative MTC and FTC, causing core power level and heat flux to increase. Positive reactivity addition is terminated on CEA insertion post-SIAS. Turbine trip causes loss of AC (LOAC), which causes reactor coolant pumps (RCPs) to coast down, minimizing core flow, lowering DNBR and maximizing failed fuel pins. Cooldown of the RCS is terminated when the affected SG blows dry and AFW flow is isolated to the ruptured steam generator.

Regulatory Guidance

- "Steam System Piping Failures Inside and Outside of Containment (PWR)", SRP 15.1.5, Rev. 2, July 1981
- "Power Levels of Nuclear Power Plants", Regulatory Guide 1.49 Rev. 1, December 1973
- "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 1.25, March 23, 1993.
- "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, March 23, 1993.

Deviations from Guidance and Clarifications

- Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, June 1, 2000).
- The power peaking factor is per the COLR/Technical Specifications and not the factor of 1.65 per Regulatory Guide 1.25.
- The failed fuel fraction is that fraction of the fuel rods whose minimum DNBR is below the design limit or that fraction of the fuel rods, which exceed the minimum enthalpy limit per the current staff approach in NEI-99-03.
- The gas gap fractions are per Regulatory Guide 1.25 or per the current staff approach in NEI-99-03: 8% I-131, 10% Kr-85, 5% other noble gases, 5% other halogens, and 12% alkali metals. For peak rod exposures <54 GWD/MTU or <62 GWD/MTU with peak rod average <6.3 kW/ft
- The MSLB creates an iodine spike in the primary RCS. The I-131 DEQ concentration in the RCS is estimated using a spiking model which assumes that the iodine release rate from the fuel rods to the RCS increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value stated in the Technical Specifications

and lasts for the duration of the accident. The current staff approach in NEI-99-03 would permit a spiking duration of 3 hours.

- An expansion model of the affected steam generator blowdown may be assumed.
- Alternate Repair Criteria may be employed. The primary-to secondary leak rate and the primary I-131 DEQ concentration may be determined from the flex methodology of DG-1074.

2.2.2. BWR Main Steam Line Break

Accident Description

The BWR main steam line break (MSLB) accident description postulates a main steam line ruptures outside containment, releasing primary coolant activity into the turbine building. Two representative conditions for the primary coolant activity concentration are evaluated: (1) a pre-accident iodine spike is assumed depicted the condition where a reactor transient occurs prior to the accident and (2) the maximum equilibrium concentration permitted for continued full power operation is assumed.

Regulatory Guidance (With Departures and Clarifications)

- "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," Regulatory Guide 1.5 (Safety Guide 5), USNRC, Rev. 3/10/71.
  - This guide focuses on evaluating offsite doses. See general guidance below on evaluating control room doses.
  - Dose conversion factors based on ICRP 30 may be used instead of ICRP 2.
  - Atmospheric dispersion factors may be calculated using Appendix D.
- "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Section 15.6.4, "Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)," USNRC, Rev. 2, July 1981.
  - This guide provides the acceptance criteria for offsite doses.
  - This guide requires two cases of iodine concentration to be evaluated, with the offsite dose acceptance criteria different for each case.
- "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Section 6.4, "Control Room Habitability System," USNRC, Rev. 2, July 1981.
  - This guide provides the acceptance criteria for control room doses.
  - The thyroid dose limit is 50 Rem instead of 30 Rem.

- "Clarification of TMI Action Plan Requirements," NUREG-0737, USNRC, November 1980.

General Guidance on Calculating Control Room Dose for MSLB

- The activity within the reactor coolant and the total amount of coolant released from the break may be determined using Regulatory Guide 1.5.
- The radiological consequences of an MSLB may be evaluated by assuming that the reactor coolant that is released from the break forms a steam cloud that migrates towards the control room at a rate of 1 meter per second.
- The effect of buoyancy on the steam cloud transport may be credited with appropriate modeling assumptions.
- The activity concentration in the cloud may be determined assuming either Gaussian or uniform distribution.
- Based on the velocity and size of the steam cloud, the length of time required for it to pass by the control room may be determined.
- For the duration that the steam cloud is passing by the control room, the activity within the cloud may be drawn into the control room via filtered and unfiltered pathways, depending on plant-specific control room HVAC system response.
- Based on the time-dependent activity inside the control room, the 30-day dose to the operator is calculated.

2.2.3. Steam Generator Tube Rupture

The description of the SGTR needs to be added

2.3. DNB Accidents

There is a subset of accidents for which the source term consists of release of the gap activity from the fuel rods that sustain departure from nucleate boiling (DNB) or breach the critical power ratio (CPR) limit and are thereby predicted to sustain cladding failure. This is referred to as "failed fuel" (versus "defective fuel which leaks and causes the iodine spiking described in the previous section), and results in the instantaneous release of the activity in the fuel rod gap and plenum to the reactor coolant system (RCS). These accidents are generally taken to be the Rod Ejection Accident (REA), the BWR Rod Drop Accident (CRDA), and the Locked Rotor Accident (LRA). These events are discussed here.

2.3.1. PWR Rod Ejection Accident (REA)

Accident Description

For the postulated control rod ejection accident, a mechanical failure of a control rod mechanism housing is assumed such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position. The resulting reactivity insertion rate may result in a power burst capable of rupturing fuel pins, melting fuel, and could breach the primary system. The power burst may cause fuel failure due to a fuel enthalpy increase above a threshold value. Two release paths to the environment are evaluated: 1) Transport from Containment, and 2) Transport from Secondary System. If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this accident are bounded then by the consequences projected for the loss-of-coolant accident (LOCA), main steam line break, and steam generator tube rupture.

Regulatory Guidance

- USNRC, "Assumptions used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors ", Regulatory Guide 1.77, Rev. 0, May 1974.

  Deviations from Guidance or Clarifications: Dose Methodology as discussed under General Outline of Control Room Habitability Analysis in this Appendix. Control Room Atmospheric Dispersion as discussed in Appendix D.

- USNRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800, Section 15.4.8 Spectrum of Rod Ejection Accidents, and its associated Appendix A, Radiological Consequences of a Control Rod Ejection Accident, Rev. 2, July 1981.

  Deviations from Guidance or Clarifications: Dose Methodology as discussed under General Outline of Control Room Habitability Analysis in this Appendix. Control Room Atmospheric Dispersion as discussed in Appendix D.

- USNRC, “Clarification of TMI Action Plan Requirements,” NUREG-0737, November 1980.
- USNRC, “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,” Regulatory Guide 1.52, Revision 2, March 1978.

- "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, 3/23/62.
- USNRC, "Laboratory Testing of Nuclear-Grade Activated Charcoal," NRC Generic Letter 99-02, June 3, 1999.
- USNRC, "Power Levels of Nuclear Power Plants", Regulatory Guide 1.49, Revision 1, December 1973.

  Deviations from Guidance and Clarifications: Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, 6/1/2000).

- USNRC, "Steam Generator Tube Integrity," Draft Regulatory Guide DG-1074, December 1998.
- USNRC, "Steam Generator Tube Rupture Analysis Deficiency," Information Notice 88-31, May 25, 1988.

2.3.2. BWR CONTROL ROD DROP ACCIDENT

Accident Description

The control rod drop accident (CRDA) is the result of a postulated event in which a highest worth control rod drops from the fully inserted or intermediate position in the core. The highest worth rod becomes decoupled from its drive mechanism. The mechanism is withdrawn but the decoupled control rod is assumed to be stuck in place. At a later moment, the control rod suddenly falls free and drops to the control rod drive position (fully withdrawn). This results in the removal of large negative reactivity from the core and results in a localized power excursion.

For large, loosely coupled cores, this would result in a highly peaked power distribution and subsequent operation of shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion.

The termination of this excursion is accomplished by automatic safety features of inherent shutdown mechanisms. Therefore, no operator action during the excursion is required.

Source Term Development

- The combination of reactor operating mode, control rod positions, core burnup, etc., that results in the largest source term is selected for evaluation.
- No allowance is made for activity decay prior to accident initiation, regardless of the reactor status for the selected case.

- The amount of radioactivity accumulated in the fuel-clad gap is assumed to be the same as that in Regulatory Guide 1.77.
- The nuclide inventory of the fraction of the fuel which reaches or exceeds the initiation temperature of fuel melting (typically 2842 C) at any time during the course of the accident is calculated and 100% of the noble gases and 50% of the iodines contained in this fraction are assumed released into the reactor coolant. NRC should be requested to review analyses that propose that fuel melting is not likely to result in significant releases prior to MSIV closure.
- Those fuel rods presumed to fail are assumed to have operated at power levels 1.5 times that of the average power of the core.
- Any nuclides released to the reactor coolant from fuel cladding failures or fuel melting are instantaneously and uniformly mixed in the reactor coolant and pressure vessel at the time of the accident.
- For conservative analysis it is assumed that 10% of the iodines and 100% of the noble gases released in the pressure vessel reach the turbine and condensers. A more realistic analysis may be performed as needed on a case-by-case basis. Such analysis accounts 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 the MSIV closure time.

Miscellaneous Input Parameters and Initial Conditions

- A coincident loss of offsite power is assumed at the time of the accident.
- The integrity of the turbine and condenser is unaffected by the rod drop accident.

Activity Transport to the Atmosphere

- All noble gases remain in a gaseous state and are available for leakage from the turbine and condensers.
- Of the iodines that reach the turbine and condensers, 90% are removed by partitioning and plateout in the turbine and condensers leaving 10% airborne and available for leakage.
- The turbine and condensers leak to the atmosphere at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate. Condenser leakage rates lower than 1% per day and shorter in duration than 24 hours will be reviewed on a case-by-case basis. Credit for condenser vacuum discharge isolation on high activity level in the steam, or credit for filtration of the condenser vacuum discharge, will also be reviewed on a case-by-case basis.
- Radiological decay during holdup in the turbine and condensers is evaluated.

Accident Duration:

24 hours unless demonstrated shorter by plant design analysis

Acceptance Criteria:

The acceptance criteria are based on the requirements of 10 CFR Part 100 as related to mitigating the radiological consequences of an accident. The plant site and dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated control rod drop accident if the calculated whole body and thyroid doses at the EAB and LPZ are well within the exposure guideline values in 10 CFR Part 100, paragraph 11. "Well within" is defined as 25% of the 10 CFR Part 100 exposure guideline values or 75 Rem for the thyroid and 6 Rem for whole body doses.

Based on past reviews by NRC staff, a control rod drop accident is expected to result in radiological consequences less than 10% of the Part 100 guideline values even with conservative assumptions. Unless unusual plant or site features are present or calculations show an unusually large amount of fuel damage, a specific calculation of the radiological consequences is not necessary. In this case a comparison of the pertinent plant and site features is sufficient to conclude that the consequences of this event meet the acceptance criteria.

Regulatory Guidance

- USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Section 15.4, Rev. 2, July 1981.

Deviations from Guidance or Clarifications: Control Room Atmospheric Dispersion as discussed in Appendix D.

- USNRC, "Power Levels of Nuclear Power Plants", Regulatory Guide 1.49, Revision 1, December 1973.

Deviations from Guidance and Clarifications: Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, 6/1/2000).

2.3.3. PWR Reactor Coolant Pump Rotor Seizure and Shaft Break

Accident Description

The accident is initiated by a seizure of the rotor or the break of the shaft of a reactor coolant pump (in PWR), which causes flow through the affected loop to be rapidly reduced. Some reverse flow may be expected through the affected loop (NUREG-800, SRP Sections 15.3.3-15.3.4). Reactor and Turbine trips occur with an assumed loss of offsite power. All the remaining reactor coolant pumps stop and cool down is performed by the operator releasing steam to the environment using the natural circulation emergency operating procedure. This is the limiting case, because with offsite power available the remaining coolant pumps would continue to operate and steam would be dumped to the condenser.

In current accident analyses for this event fuel failure is generally assumed to occur at the onset of DNB, even though the rods may be in a film boiling condition for a very short period of time. In fact, it is unlikely that fuel failure will occur. More advanced analysis of this event could demonstrate non-limiting results for CRH evaluations.

Some fuel damage may be expected when the reactor is at power (HFP condition) due to the loss or reduction of coolant flow in the affected loop. A reactor core design specific T&H analysis will provide the percentage of fuel failure resulting from some of the fuel rods going into DNB. Generally, the fuel damage value varies in the ranges of 0% to 15%. However, statistical DNB analysis methods may show no rods going into DNB and no fuel failure. For rods going into DNB for a short duration, (approximately 10 seconds) fuel failure may not occur. Alternative methods of evaluation, such as utilizing the fuel rod enthalpy as a measure of fuel failure would reduce the conservatism inherent in the use of the DNB or CPR methods for these cases.

No increase in the leakage of the primary coolant to the secondary side is expected. However, a larger amount of activity may be transported to the secondary side via any preexisting leaks in the steam generators.

Activity is released to the environment through safety valves and/or power operator relief valves (PORV) until the affected loop is either isolated in the case of a stuck open PORV or until the primary system is cooled down. Unaffected loop's safety valves and/or PORVs continue to function until the reactor is cooled down. This is consistent with the loss of offsite power assumption.

The sequence of events must consider any time delays prior to and after protective system actuation. Briefly, the reactor is at power when the event occurs at t=0.0 seconds. Power to the other pumps is lost immediately or within a few seconds. The PORVs on the two unaffected steam generators open within a few seconds of the event and the PORV on the steam generator in the affected loop opens within one minute.

Two cases should be considered: one where the PORV sticks open and a second case where the PORV cycles between open and closed. The cycling PORV case can result in the bounding analysis because the release can occur for a longer period of time (up to 8 hours, use a lower value if justified).

The Westinghouse Owners Group has suggested that based on the low probability of both a locked rotor combined with steam generator tube uncovery, no tube uncovery should be assumed (see WCAP-13132).

Source Term

Coolant activities in the primary and the secondary systems must be determined for the dose analysis. Generally, Technical Specifications limit the steady state primary coolant specific activity to 1.0 micro-Ci/gm dose equivalent I-131 and the secondary coolant specific activity to 0.1 micro-Ci/gm dose equivalent I-131. In addition, the Technical Specifications allow the activity in the primary system to spike to 60 micro-Ci/gm dose equivalent I-131 or to some other plant specific value (pre-accident spike factor) for a short period of time. During rapid power and pressure changes coolant activity increases as the result of iodine spiking.

As discussed above, an additional source of activity in the primary coolant is the release from additional fuel failure. This activity can be estimated by multiplying the core inventory by the gap fraction (10% or less) and then by the fraction of rod failure during the event. The gap activity consists of 10% of the core inventory of iodine and noble gas and 30% of the Kr-85. (This distribution of released isotopes is the same as is used for the fuel handling accident).

Generally, the primary coolant activities are given in the UFSAR based on the assumption of 1% failed fuel and 102% power as required by the SRP. The primary to secondary leak rate is generally controlled by Technical Specification and this leak rate should be modeled.

Additional Modeling Assumptions

For simplicity of modeling, releases can be treated as being identical through all steam generators for the entire release period. A single liquid volume and a single steam volume can be used provided it represents the release of liquids and steam through all the steam generators. The release should be assumed to occur for a period of 8 hours by which time the coolant system temperature will be decreased to 350 F (normally cool down is achieved in 4 hour). At this temperature, RHR is activated and the release to the environment through the steam generator PORV ceases.

Accident Duration

Affected generator with stuck PORV, release stops within 30 minutes, Cycling release stopped in 8 hours or less.

Regulatory Guidance

- USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Section 15.4, Rev. 2, July 1981.

  Deviations from Guidance or Clarifications: Control Room Atmospheric Dispersion as discussed in Appendix D.

- USNRC, "Power Levels of Nuclear Power Plants", Regulatory Guide 1.49, Revision 1, December 1973.

  Deviations from Guidance and Clarifications: Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49. A factor of 1.01 may be employed crediting improved thermal power measurement accuracy with NRC approval of an exemption of 10CFR50 Appendix K (See Federal Register, Vol. 65 No. 106, 6/1/2000).

References

- NUREG-800 SRP Section 15.3.3-15.3
- NUREG-800 SRP Section 6.4
- NUREG/CR-6331, Rev. 1, "Atmospheric Relative Concentrations in Building Wakes, ARCON96, USNC 1997.
- 10CFR50 Appendix A GDC 19
- 10 CFR 100
- Reg. Guide 1.25
- Reg. Guide 1.145
- Reg. Guide 1.52
- Reg. Guide 1.4

2.4. [2.4] Fuel Handling Accident

Text to be developed

3. Evaluation and Treatment of Uncertainties in Dose Analyses

3.1. Establishing Appropriate Levels of Conservatism

An acceptable level of overall conservatism (e.g., a 95th percentile Control Room dose) should be established. All contributing factors should be examined to ensure that, at least approximately, that level of conservatism is being achieved but not greatly exceeded. This may be more important for Control Room analysis than for offsite dose analysis because of the greater number of steps involved in the Control Room dose analysis and, therefore, the greater potential for excessive conservatism.

3.2. Statistical Treatment of Uncertainties

Description of Potential Monte Carlo evaluations of key uncertainties in input and modeling for CR dose analysis [To be developed]

APPENDIX H

TOXIC GAS ASSESSMENTS

1. PURPOSE

This appendix provides guidance to assess control room habitability during and after a postulated external release of hazardous chemicals. An existing toxic gas evaluation should be revised if the assumed inleakage value is found to be non-conservative or if a new significant source of hazardous chemical is identified in the vicinity if the plant. In addition this Appendix provides guidance to those plants where a periodic reassessment of toxic gas challenges is warranted.

2. SCOPE

This appendix applies to the external release of hazardous chemicals from mobile or stationary sources, offsite or onsite. It does not consider the explosive hazard of these chemicals, which is considered beyond the scope of this appendix.

3. REGULATORY BASIS

Criterion 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. Criterion 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 (Reference 6.1).[6] This appendix also provides guidance beyond that contained in

[6] Regulatory Guide 1.95 (Reference 6.2) also discusses protecting control room operators against accidental chlorine releases. The guidance related to chlorine releases provided in Regulatory Guide 1.95 is not presented in this appendix; the reader is encouraged to refer to Regulatory Guide 1.95 for chlorine-specific concerns.

Regulatory Guide 1.78 in the areas of specifying toxicity limits, identifying sources of onsite and offsite hazardous materials, determining hazardous chemical release characteristics, and applying updated atmospheric dispersion modeling techniques, including dense gas atmospheric dispersion models.[7] Licensees following the guidance of this appendix may use the:

- methodology that currently serves as their licensing basis,
- guidance presented in Regulatory Guide 1.78 as supplemented by this appendix, or
- other regulatory guidance that is subsequently published by the NRC.

4.1 IDENTIFYING HAZARDOUS MATERIALS

4.1.1 OFF-SITE

Two federal laws were specifically designed to provide information regarding hazardous chemicals at industrial facilities. The EPA, as do 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 Emergency Planning and Community Right-to-Know Act (EPCRA) and the Clean Air Act's Risk Management Program (RMP) both require facilities to report on hazardous chemicals they store or handle, and both provide for public access. 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's important to remember that there are data limitations. 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. The same agencies may have information on the transport of hazardous materials via railways. If not, the railways should be contacted directly. Information on river, Great Lake, and coastal marine traffic can be obtained from the U.S. Coast Guard.

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[7] Some of the guidance presented here that is related to Regulatory Guide 1.78 is contained in NUREG/CR-6624 (Reference 6.3)

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 ONSITE

A facility's EPRCA and RMP reporting information, if required, 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 EPRCA 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 Immediately Dangerous to Life and Health (IDLH) exposure levels published by the National Institute for Occupational Safety and Health (NIOSH) (References 6.4 and 6.5).

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 2-minute exposure limits. 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 onsite 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% 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 inleakage characteristics of the control room, and the applicable toxicity limits.

Section 5.2.1 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 5.2.2 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 onsite 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 and stable meteorological conditions.[8]
- *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 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

For each chemical considered, the value of importance is the maximum concentration that can be tolerated for two minutes without physical incapacitation of an average human (i.e., severe coughing, eye burn, or severe skin irritation). The two-minute

8 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.

criterion is based on the time a control room operator is expected to take to don respirator and protective clothing. As stated in Section 5.1, the two-minute toxicity limit can be based on either the toxicity limits listed in Table C-1 of Regulatory Guide 1.78 or the Immediately Dangerous to Life and Health (IDLH) exposure limits formulated by the National Institute for Occupational Safety and Health (NIOSH).

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, it is expected that compensating measures will be implemented.[9] 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.[10] 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 to prolonged exposure at the two-minute toxicity limit levels.

There are several aspects which 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.

- 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 of the hazardous chemical to be considered 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 5.2.1; 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 5.2.1; or 3) the largest container stored onsite (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 onsite source failing the screening criteria outlined in Section 5.2.1 should be considered.

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[9] 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 (Reference 6.6).

[10] The American Industrial Hygiene Association (AIHA) has established odor thresholds for a number of toxic chemicals (Reference 6.7). Some of these data are presented in NUREG/CR-6624 (Reference 6.3).

- 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 which, 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 should be consulted for more specific details concerning performing evaluations of control room habitability for potential toxic gas releases.

Regulatory Guide 1.78 suggests utilizing algorithms presented in its Appendix B for performing atmospheric dispersion modeling for instantaneous (puff) releases and algorithms presented in Regulatory Guide 1.3 and 1.4 for performing atmospheric dispersion modeling for continuous releases (References 6.8 and 6.9, respectively). Other options for performing atmospheric dispersion modeling analyses include utilizing Murphy and Campe (Reference 6.10) for releases near the control room (within 100m or so) and Regulatory Guide 1.145 (Reference 6.11) for releases further from the control room.

NUREG-0570 (Reference 6.12) 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 recently sponsored the development of a computer code system for evaluating control room habitability called HABIT (References 6.13 and 6.14). 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 of 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% 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% 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 6.17). The latter reference also provides guidance on how to execute several generally available dense gas atmospheric dispersion models.

5 References

5.1 Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," U.S. Atomic Energy Commission, June 1974.
5.2 Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," U.S. Nuclear Regulatory Commission, January 1977.
5.3 LB Sasser et al, "Recommendations for Revision of Regulatory Guide 1.78," NUREG/CR-6624, Pacific Northwest National Laboratory, November 1999.
5.4 H.R. Ludwig, S.G. Cairelli, and J.J. Whalen, "Documentation for Immediately Dangerous to Life or Health Concentrations (IDLH)," National Institute for Occupational Safety and Health, 1994.
5.5 National Institute for Occupational Safety and Health, "NIOSH Pocket Guide to Chemical Hazards," U.S. Government Printing Office, 1997.
5.6 NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission, July 1981.
5.7 American Industrial Hygiene Association (AIHA), "Odor Thresholds for Chemicals with Established Occupational Health Standards," AIHA, 1989.
5.8 Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors" U.S. Nuclear Regulatory Commission, 1974.

5.9 Regulatory Guide 1.4, Rev. 1, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors" U.S. Nuclear Regulatory Commission, 1974.
5.10 KG Murphy and KM 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.
5.11 Regulatory Guide 1.145, Revision 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, 1982.
5.12 J Wing, "Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release," NUREG-0570, U.S. Nuclear Regulatory Commission, June 1979.
5.13 SA Stage, "Computer Codes for Evaluation of Control Room Habitability (HABIT)," NUREG/CR-6210, Pacific Northwest National Laboratory, June 1996.
5.14 JV Ramsdell and SA Stage, "Computer Codes for Evaluation of Control Room Habitability (HABIT)," NUREG/CR-6210 Supplement 1, Pacific Northwest National Laboratory, September 1998.
5.15 EPA-454/R-92-024, "Workbook of Screening Techniques for Assessing Impacts of Toxic Air Pollutants (Revised)," U.S. Environmental Protection Agency, December 1992.
5.16 EPA 550-B-99-009, "Risk Management Program Guidance for Offsite Consequence Analysis", U.S. Environmental Protection Agency, December 1992.
5.17 EPA-454/R-93-002, "Guidance on the Application of Refined Dispersion Models to Hazardous/Toxic Air Pollutant Releases," April 1993.

APPENDIX I

SYSTEM ASSESSMENT

1 PURPOSE

The purpose of this appendix is to provide guidance for utility personnel in performance of walkdowns and inspections of the control room envelope and associated ventilation systems to identify potential vulnerabilities to leakage.

2 SCOPE

This appendix provides guidelines to assist personnel in the performance of walkdown activities of the control room envelope and the associated ventilation systems with the intended purpose of identifying potential vulnerabilities to inleakage into the Control Room envelope.

This appendix does not provide guidance for minimizing these vulnerabilities. Appendix K provides the guidance for sealing or minimizing these vulnerabilities.

3 WALKDOWN METHODOLOGY

3.1 IDENTIFY THE BOUNDARY

The function of this section is to ensure that the user has a good understanding of the boundaries and performance requirements for the envelope and the ventilation system(s). This process would be performed similar to the following:

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

5.1.2 On the drawings highlight the following:

- Boundaries of the Envelope
- Boundaries of the Ventilation System(s) that serve the Envelope. Identify portions of the ventilation system(s) that are physically located outside of 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. This may require more than one set of drawings if the system response is different for different types of events.

- Identify Ventilation System(s) that traverse the Envelope boundary. Highlight and label on the drawings the routing of other ventilation systems that traverse the envelope.
- Establish the design performance parameters for the system for the different accident types (radiological, toxic gas). These parameters include but are not limited to differential pressures, make up and recirculation flow rates, duct static pressures, and filter differential pressures. The purpose of this activity is to allow identification of portions of the control room envelope that are at lower pressure than that of surrounding areas or traversing HVAC systems. 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 Identify Operating Configurations

Control room in-leakage must be measured with affected systems in their accident configuration. See Appendix J Section 5.2 for additional guidance with regards to operating modes for the systems. The walk-down should confirm that all components can be configured in their accident modes.

3.2.1 Consider the Accident(s)

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 events (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 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 SI event vs. a Control Room high radiation event (non-SI event).

3.2.2 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.

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.

3.2.3 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 and the fan shafts are not sealed.

3.2.4 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 the potential location for in-leakage.

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

3.3.1 VISUAL EXAMINATION

During the walkdown, the user(s) 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, below.

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. If smoke blows into the boundary, this is considered to be in-leakage and affects both radiological and toxic gas assessments. If smoke blows out of the boundary, this is considered to be out-leakage. Out-leakage may affect the ability of a positive pressure system to sufficiently pressurize the envelope. Out-leakage requires additional make-up air to maintain the positive pressure; even though this air is usually filtered, it still affects radiological and toxic gas assessments.

3.3.2 SYSTEM FLOW MEASUREMENTS

Airflow rates should be measured to ensure that the system flow rates are as expected for the various configurations. 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.

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.

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).

3.4 SPECIFIC CONSIDERATIONS (THINGS TO LOOK FOR)

Table I-1 provides a list of items to consider when evaluating potential vulnerabilities to in-leakage. The items in the table are applicable to several different potential system and envelope configurations. Depending on system and envelope configuration not all of these may be applicable to any given facility. The user of this information should be careful not to use this as a complete listing for their facility, but apply it as guidance for the types of vulnerabilities to look for. These vulnerabilities coupled with the positive pressure test may be used to justify a component test per Appendix J.

The additional description below is to aid the user in the use of the table.

3.4.1 CONTROL ROOM VENTILATION SYSTEM

For portions of ventilation systems located outside of the envelope:

- In-leakage can occur into portions of the ventilation system that are located outside of the envelope if portions of these systems (e.g., return ducting) are at a negative pressure relative to the area(s) they are routed through.
- Ventilation ducting (commercial, pocket lock, non-seal welded, non-bolted connections, etc.) can be 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.
- 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.

For portions of ventilation systems located inside the envelope:

- Portions of pressurization ductwork upstream of the filter and within the CRE 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 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 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.

3.4.4 Doors in Control Room Envelope Boundary

Door seals can be a potential significant source of in-leakage. Previous 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 verification 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. Louvered dampers are discussed here because they 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 of 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 in the past 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 case, it may be easier to verify the boundary by looking at the other side.

4 Documentation

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

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

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 and would not provide undesirable differential pressures across the envelope boundary. Determine that system air flow rates 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. Consider the following types of items: Ductwork Equipment housings System penetrations such as 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.
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.

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 seals inside the conduit and between the conduit and the wall. Determine that ventilation ducting penetrations and dampers are properly sealed. Determine that drains (floor or equipment) have loop seals or check valves. 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; for example, concrete anchors through block walls which are not sealed.
Envelope Doors (Section 3.4.4)	Determine that there are no defects in doors. Determine that door seals are not cracked, are not missing, and have proper fit. Determine that doors are properly compressed against the door seals. 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, and have proper fit. 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 instrument 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 conditions (check concrete, block, expansion joints, etc.)

Vulnerability	Determining Vulnerability Significance
Control Room Ventilation System Integrity (Section 5.4.1)	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. Consider the following types of items: Ductwork Equipment housings System penetrations such as electrical and instrumentation Accesses such as doors or hatches Fan Shaft (AHU, Recirculation fan, etc.) Determine if portions of the pressurization ducting between the envelope boundary and the filter is 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.
Control Room Ventilation System Operation (Section 5.3.2)	Determine that ventilation systems are properly balanced and would not provide undesirable differential pressures across the envelope boundary. Determine that system air flow rates are as expected
Other Ventilation Systems (Section 5.4.2)	Determine if other system ducting are routed through the envelope when the control room is isolated. Determine the post-accident pressure in the ducting relative to the pressure in the envelope (consider both the effects of this ducting as a means of inleakage or out-leakage). If the ducting is isolated, consider the potential for damper leakage. Determine the integrity of this ducting.
Penetrations in the Envelope Boundary (Section 5.4.3)	Determine that wall, floor and ceiling penetrations (i.e., conduits, electrical cable trays, etc.) are properly sealed. Check seals inside the conduit and between the conduit and the wall. Determine that ventilation ducting penetrations and dampers are properly sealed. Determine that drains (floor or equipment) have loop seals or check valves. If used, verify that the check valve is appropriate for this application. Determine if there are other types of penetrations that can provide potential leakage pathways; for example, concrete anchors through block walls which are not sealed,
Envelope Doors (Section 5.4.4)	Determine that there are no defects in doors. Determine that door seals are not cracked, are not missing seals and have proper fitting seals. Determine that doors are properly compressed or fitting against the door seals. Determine that door latches are functioning properly to maintain the door securely closed. Determine that doorframes are properly sealed.

Isolation Dampers (Section 5.4.5)	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 5.4.6)	Determine if there are instrument or service air lines that enter the envelope boundary and could provide potential unfiltered air sources due to leakage, operational venting of air operated components, etc. Consider other equipment operations providing a mechanism for air inleakage such as Radiation Monitors that are located outside the envelope and draw a sample from within the envelope.
General Boundary Construction (Section 5.4.7)	Determine that the general envelope boundary is in good conditions (check concrete, block, expansion joints, etc.)

APPENDIX J

TESTING PROGRAM

1 PURPOSE

This appendix provides guidance on the development of a testing program to verify control room boundary integrity in support of a demonstration that the control room habitability system conforms to the plant licensing/design basis. The appendix also provides guidance on preparing for an in-leakage test.

2 SCOPE

The guidance in 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. Guidance on acceptable testing options, preparation for testing, performance of testing, and test frequency is provided. This guidance is intended to aid plant personnel in the development of a plant specific procedure for testing.

Attributes of an acceptable test program

- The test must be comprehensive.
- System testing must be conducted with systems and components in their accident configuration lineups.
- Testing methods should be tied to a recognized industry standard
- Component testing must be conducted in a manner that reflects accident configuration leakage

3 REGULATORY BASIS

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

4 TEST/TEST PROGRAM DEVELOPMENT

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

4.1 PREREQUISITES TO DEVELOPING TEST/TEST PROGRAM

4.1.1 Prior to developing a test or test program an assessment of the control room boundary should be performed in accordance with Appendix I of this document. This includes the walkdown portion of the appendix plus any sealing/refurbishment/repairs needed as identified in the assessment.

4.1.2 Prior to conducting testing, 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 Leakage for Continued Operation, and Maximum Allowable Toxic Gas In-leakage. (Appendix C provides analytical guidance on the calculations, Appendix F has guidance on compensatory measures).

4.1.3 HVAC systems (including adjacent spaces HVAC systems) should be properly aligned and balanced to meet required air flows and pressures. This information is verified in Appendix I.

4.1.4 The impact on other plant activities should be assessed. The ingress and egress of the control room boundary may need to be limited during the test.

4.1.5 Determine acceptance criteria.

4.1.6 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.

4.2 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 would not be required.

The system modes for testing shall be documented along with the basis for the system mode tested.

4.3 DETERMINE METHOD OF BASELINE TESTING

In-leakage baseline values must be performed on 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 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 should depend on the vulnerability of the plant to in-leakage. For plants where a large number of vulnerabilities are identified or testing a specific component vulnerability is not feasible, an Integrated Tracer Gas Test would most likely be the best test method. On the other hand, 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 the best test method. The plant may perform an economic evaluation of the different test methods being considered. It is possible that an integrated tracer gas test may be less expensive to perform then a component test.

For CREs that can tolerate large amounts of unfiltered in-leakage, Alternate Tests (i.e., flow measurements) may be acceptable.

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 4.3.1, 4.3.2 and 4.3.3 provide additional information that will assist in determining the type of testing to be performed.

4.3.1 INTEGRATED TRACER GAS TESTING

For control room envelopes that are not positive to all adjacent areas or have numerous vulnerabilities to in-leakage, Integrated 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 leakage of the control room envelope by an integrated test. The measurement of the concentration, and sometimes the volume rate of the tracer gas that is injected into the zone, allows calculation of the volume rate of outgoing air from the zone. The volume of incoming air can be inferred from these measurements. This test method does not determine the in-leakage contribution of individual components.

Integrated Tracer Gas testing uses American Society for Testing and Materials (ASTM) Standard E 741, "Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution". Procedures for tracer gas dilution include concentration decay, constant injection, and constant concentration.

Consider the following items when performing a test:

- This test is heavily dependent upon ensuring tracer gas concentration is uniform 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 critical to the success of this test method.
- Determination of the net volume of the control room envelope is equally 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.
- The constant injection technique has generally proven the most effective method.
- 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 this 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.

Integrated tracer gas testing has proven useful for measuring in-leakage into the control room envelope. This testing method is heavily dependent on proper techniques and may be difficult to perform for complex control room designs.

4.3.2 COMPONENT TESTING

For positive pressure CRE designs with a small number of vulnerabilities to in-leakage, Component Testing may be the most straightforward and 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 is located within the control room envelope
- Minimal non-control room ventilation ducting or air system piping penetrate the control room envelope.

- Ventilation ducting is of the seam welded design and is in good material condition

This method is heavily dependent upon a thorough assessment of the control room envelope boundary and ventilation systems to ensure all potential vulnerabilities are properly evaluated. The use of independent peer industry personnel on the assessment team is recommended. Thorough documentation of the assessment results is critical 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 J-1 provides information on some industry testing standards that can be used for a component test.

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

- First, the control room envelope differential pressures must be measured in sufficient 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 I assessment. The sum of the individual component in-leakage values will become the total in-leakage rate.

Component testing must result in identification of a total CRE in-leakage rate in CFM. This value is the sum of the leakage of individual components. A comprehensive assessment must accompany this testing to assure that all potential in-leakage pathways are tested. Each component test should meet an acceptable national standard.

Component Testing must be conducted in a manner that reflects accident configuration leakage. The test differential pressure across the component must reflect an equivalent differential pressure to that which the component would sense in an accident condition. The effect of HVAC systems in adjacent areas that may not operate in accident conditions must be accounted for when establishing Component Test conditions.

This test method can determine total in-leakage. It may be less complex than the integrated tracer gas test for systems with few vulnerabilities and subsequently fewer components to test. Component Testing should be within the capability of the plant staff.

The main disadvantage of this test method is that the assessment may miss an in-leakage pathway that subsequently would not be tested. This test method is not considered applicable to control room designs that are not pressurized in the emergency mode or where many vulnerabilities to in-leakage may exist. Therefore it is imperative that a thorough evaluation per Appendix I is performed and that accurate differential pressure measurements are made.

4.3.3 ALTERNATE TEST METHODS

Each licensee 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 system envelope.
- Performed in accordance with an industry test standard.
- 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 J-1 for methods that may be considered to develop alternative tests. Note that a combination of methods may be necessary to produce an overall in-leakage value in cfm for the entire envelope.

4.4 PERFORMANCE OF BASELINE TESTING

Based on the determination made in section 4.3 either section 4.4.1 (tracer gas) or 4.4.2 (component) may be utilized. If an alternate test method is chosen then the utility should establish the guidance related to the alternate test.

4.4.1 INTEGRATED TRACER GAS TESTING

The industry standard currently being used for a tracer gas test to determine in-leakage is ASTM E741. This test method is heavily dependent on proper technique and may be difficult to perform for complex control room designs. It is beyond the scope of NEI 99-03 to provide a detailed procedure in applying ASTM E741; however guidance is presented in preparing and conducting the test.

Following completion of Sections 4.1, 4.2. and 4.3 above; there is a sequence of three steps recommended in preparing for and conducting the integrated tracer gas test. These steps are:

- determine if the test is to be performed in-house or using a contractor
- perform a walk down of the CRE (including sealing of potential leak paths identified in Appendix I); and
- perform the test.

4.4.1.1 PRELIMINARY ACTIONS

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 onsite.
b. Material Safety Data Sheets should be obtained for the tracer gas for incorporation/approval by the site's materials control program.
c. Determine the net volume of the CRE. 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 (M&TE).
   - 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 (onsite) 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 walk-down
   - Walk-down 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

4.4.1.2 PERFORMANCE OF THE TRACER GAS TEST

The test is only a snapshot of the CRE integrity. The leakage will vary due to the time of year, wind conditions, temperature, and pressurization flows. In order to obtain trendable test results, the test conditions should correspond to the analysis of the system to the greatest extent possible.

The minimum time recommended to perform the test is 2*Tau, where Tau is the reciprocal of the air exchange rate in terms of total contained volume changes per unit time. The optimal test time is twice the minimum time.

Key factors effecting accurate testing are:

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

Additional considerations for performing an effective test are:

- Follow all appropriate Technical Specification LCO 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. 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
- Include requirement to limit door openings/closings during test
- Perform testing in accordance with plant procedures
- Retest as necessary.

4.4.2 COMPONENT TESTING

This testing is dependent upon the premise that the CRE is at a positive pressure to all adjacent areas. In order to credit Component Testing for determining unfiltered in-leakage, this premise must be validated by testing. In this respect, the differential pressure measurements collected below are critical. These differential pressure measurements are used to show 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 J-2 for more discussion of the components and testing of those components.

4.4.2.1 DIFFERENTIAL PRESSURE 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.

Fluids (in this case air) flow 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 very helpful 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.

For a neutral pressure envelope design (note that this may be the case for a plant designed for positive pressure to radiation but neutral for toxic gas), leakage through the envelope boundaries can be either in or out, depending on the direction of the differential pressure. In-leakage is obviously a concern. Excessive out leakage from the envelope should be minimized as this must be off set by in-leakage through other boundaries and/or high makeup airflow rates.

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. For adjacent areas inside of a building a positive pressure of 0.05 inches water gage is recommended as this pressure is sufficiently high enough to allow accurate measurements. Precision digital barometers can be used. Barometers of accuracy of +/-0.03 inches of water exist in the industry. Precision digital manometers capable of sensing pressure changes of 0.0001 psi also exist within the industry. 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 Appendix J, Section 4.2.
- 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.
- 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 different 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.

If it is discovered that adjacent area(s) are at a higher pressure than the pressure inside the envelope, actions could possibly be taken to reduce the pressure in the adjacent area. Things to consider are ventilation system operating configurations, securing fans (if feasible) and providing pressure relief paths. This is addressed in more detail in Appendix K

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 chosen.

4.4.2.2 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 4.4.2.1 and Table J-2 to make this determination. Each vulnerability (i.e., component) identified in Appendix I 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.

4.4.2.3 SELECT TEST METHOD FOR THE COMPONENT

Available methods for testing the leak tightness of components11 are provided in Table J-2.

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

4.4.2.4 PERFORM THE APPLICABLE TEST

- Perform each test as identified in 4.4.2.3.
- Record the leakage measurements made[12].
- Sum all the leakage measurements. This is the total unfiltered in-leakage.

4.5 TEST RESULTS

- Document the components to be tested as identified in section 4.4.2.2; 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 limits; compensatory measures may need to be taken to maintain the control room ventilation system operable until

[11] 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”.

[12] For control room envelopes that can tolerate large amounts of unfiltered in-leakage, flow measurements are acceptable provided the measured considers instrument error.

permanent resolution is achieved (See Section 8.0 and Appendix F for guidance). In-leakage values that result is doses greater than that currently reported in the UFSAR will require evaluation per the plant's Corrective Action Program.

5.0 DOCUMENTATION

Appendix J-3 delineates documentation requirements.

Table J-1
Testing Options

Type of Test	Standard	Advantages	Disadvantages	Performed with systems in their accident configuration	Accuracy	Quantitative	Comments
Tracer Gas (SF6)	**ASTM E741**	**History of use within the industry 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%**	**Yes**	**1. 3.**
Pressure Test (Blower Door)	**ASTM E779**	**1. Performed under pos. or neg. press. 2. Requires CRHVAC shutdown.**	**1. Req. CRHVAC shutdown 2. Tests from inside out 3. No leak location identified 4. Impact on OPS 5. Wind Effects 6. Seal supply and return duct**	**No. CRHVAC is shutdown**	**± 5%**	**Yes**	**2.**
Leak Detection	**ASTM E1186**	**1. Identifies location 2. Inexpensive 3. No effect on OPS**		**Yes.**	**N/A. This item identifies leaks but cannot accurately quantify leaks.**	**No.**	
Component Test	**ASTM E779 ASTM E1186 ASTM E741 ASME N510 ASME AG-1**	**1. Low Cost 2. Low impact on OPS 3. Identifies leak location**	**1. Requires isolation of individual components**	**Section by Section**	**Test Dependent**	**Yes**	

Comments:

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. The increase in uncertainty depends on how the flow is measured.

2. Accuracy depends on how the flow measurement is made.

3. Testing developed by Brook Haven National Labs (BNL) 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 (website http:www.bnl.gov/ecd.htm). WEBSITE current as of 9/3/00.

Table J-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 inleakage as the leakage would be out of the CRE.	Not required as positive pressure precludes inleakage.	NA
CRE Walls	The positive pressure measurements of the CRE would show that this vulnerability would not exhibit inleakage as the leakage would be out of the CRE.	Not required as positive pressure precludes inleakage.	NA
CRE Floor	The positive pressure measurements of the CRE would show that this vulnerability would not exhibit inleakage as the leakage would be out of the CRE.	Not required as positive pressure precludes inleakage.	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 inleakage at 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 inleakage.	NA
CRE Doors	The positive pressure measurements of the CRE would show that this vulnerability would not exhibit inleakage as the leakage would be out of the CRE.	Not required as positive pressure precludes inleakage.	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 indicated passing through the open conduits then an integrated tracer gas test may be required.	Not required provided that the conduits, etc. are properly sealed internally.	NA, otherwise use smoke pencils. See discussion.
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. This is applicable to both HVAC ducting and filter system ducting. Any ducting and/or housings under a negative pressure is a potential source for inleakage. Access doors,	Required	See Acceptable Method this table

	hatches, instrument lines, drain lines (should have loop seals to prevent leakage), damper and fan shafts,		
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 inleakage	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.		Required	See Acceptable Method this table
Other systems	Radiation monitors and pneumatic air airlines may be a source of inleakge. These systems should be reviewed for leakage. Constant bleed air regulators can be a source of unfiltered inleakage 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 J-3
Appendix J
Critical Documentation Requirements

Section	Type of Test Selected*		
	Tracer Gas	Component	Alternate
5.1.5 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
5.4.1.2 Components selected to be tested	NA	Yes	NA
5.5 Inleakage values measured for each mode selected in 5.2	Yes	Yes	Yes
6.0 If periodic testing required then Basis for periodic test; type of periodic test; and frequency of test or basis for why periodic test not required	Yes	Yes	Yes

NA - Not Applicable
* - Only one type of test should be selected for toxic gas and one type for radiation. If the plant response to both events is the same then only one test that covers both toxic gas and radiation may be performed.

APPENDIX K

CONTROL ROOM ENVELOPE SEALING PROGRAM

1. PURPOSE

The purpose of a 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. DISCUSSION

The integrity of the CRE is important for two reasons.

- The amount of inleakage (filtered and unfiltered) may significantly affect the post-accident radiological dose to the operators. The safety analyses assume a value for inleakage during and following radiological and toxic gas releases. The CRE and associated mechanical components must be able to maintain the inleakage so as to meet these limits.
- For many plants, the CRE is pressurized post-accident using emergency fans. If the CRE integrity degrades enough, the fan capacity may be insufficient to maintain the design pressure.

Therefore, the allowable leakage, and the importance of a specific penetration seal, will depend on whether it primarily seals against inflow or outflow in the event of an accident. A CRH assessment, as outlined in Appendix I, will provide guidance in this area. For example, if a CRE is pressurized following a DBA, , minor leakage to adjacent areas at lower pressures will be <u>out</u> of the CRE and thus will not increase the operator dose. However, a leaking outside air damper in a suction line to a recirculation AHU located outside the CRE will result in leakage <u>into</u> the CRE. Depending on the presence and location of charcoal filters, this inleakage may be unfiltered, and could raise operator dose.

3. CRE BARRIER CONTROL

Control of the CRE pressure boundary should be maintained at all times (see Appendix L of NEI 99-03). 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.

4. SEALING PROGRAM DEVELOPMENT

A CRE assessment, as outlined in Appendix I, 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 penetration.

The following is a list of typical penetrations and/or items that may have penetrations that would allow inleakage.

Doors
Door seals
Isolation Dampers / Shafts and gaps
Fire Dampers
Gaps (required for fire damper thermal expansion) around Fire Dampers
CRE walls/ceilings/floors
Gaps at building wall/floor/ceiling intersections
Ducting traversing CRE and at higher pressure
CR pressure boundary ducting outside CRE
Duct penetrations
Duct expansion joints
Conduit penetrations
Conduits
Cable trays
Instrument air lines supplying CR pneumatic components
Other instrument lines
AHU drains
AHU housing
Filter housing/drains
Fan housing/shaft
Duct access panels

Basic guidelines for inspection are listed below; 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.

4.1 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.

4.2 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, seat 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 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.

4.3 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.

4.4 Ducting, duct penetrations, expansion joints

Welded ducting is preferable. For other types, all seams and connections should be sealed with an approved sealant, such as RTV or hardcast, and tested for leaktightness (Snoop or pressure decay methods). Duct penetrations should be 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, 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.

4.5 ELECTRICAL CONDUITS, CONDUIT PENETRATIONS, 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 which may not be readily apparent during a normal visual walkdown or inspection.

4.6 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.

4.7 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.

5. 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.

- Major equipment (AHUs, filters, dampers, etc) and long duct runs located outside the envelope significantly increase the potential for unfiltered inleakage, and the effort required to detect and measure the inleakage.
- 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.
- 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.
- Careful flow balance testing may be required to resolve this problem.
- Maintaining CRE internal doors open, 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 offsite power.
- From a CRE perspective, an accident without a LOOP may actually be worse due to continued operation of none ventilation systems. 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.

6.0 REFERENCES

6.1 ANSI N510 Testing of Nuclear Air Cleaning Systems

6.2 ASME AG-1 Code on Nuclear Air and Gas Treatment

6.3 WASH-1234 ESF Air Cleaning Systems for Commercial LW-Cooled Nuclear Power Plants

6.4 ERDA 76-21 Nuclear Air Cleaning Handbook – Design, Construction, and Testing of High-Efficiency Air Cleaning Systems for Nuclear Application

6.5 NHUG Draft Guidance on Breach Control

6.6 SMACNA
HVAC Duct Construction Standard – Metal and Flexible
HVAC Air Duct Leakage Test Manual
Fire, Smoke, and Radiation Damper Installation Guide for HVAC Systems
Technical Paper on Duct Leakage

APPENDIX L

CONTROL ROOM ENVELOPE BOUNDARY CONTROL PROGRAM

1 PURPOSE

This appendix provides guidance to control breaches of the CRE and may be used to develop plant specific procedures. Scope

A boundary control program should include activities that breach the CRE such as:

- The creation of a new penetration or 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 which also serve as the CRE boundary.
- Any piping system breach (e.g., valves, pumps, or pipes) which creates an air flow path through the CRE boundary.
- The removal of equipment and/or floor drain plugs from the CRE boundary.

2 DISCUSSION

The physical CRE boundary is a fundamental element of CRE integrity. It is important to control breaching the CRE boundary to ensure that the design is maintained such that the accident analyses remain valid. 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, to monitor active breaches and to 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. Additionally, the maximum pressurization airflow rate allowed by the accident analyses may be adversely affected.

3 PROCESS

3.1 IMPACT EVALUATION

Prior to breaching the CRE boundary, the activity should be evaluated for the impact on CR habitability. This evaluation should consider as a minimum, the breach size and the ability to rapidly restore the boundary. The impact on fire boundaries, tornado protection boundaries, security boundaries, etc. should also be considered when opening up a boundary.

3.1.1 BREACH SIZE

Evaluate the effect the breach has on in-leakage margin, pressurization flow rate, and required dp 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 differential pressure across the boundary that will be breached. The third step is to calculate the maximum breach size using the allowable in-leakage and DP 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 pretest should be performed to verify that the breach size does not adversely impact the CRE dp and pressurization air flow requirements.

If it can be demonstrated that the duration that the breach will be open will not 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 will be necessary. These compensatory measures may need a 10 CFR 50.59 evaluation.

3.1.2 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.

3.2 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. Some plants may accomplish this via a breach permit tracking system while others may control the number of work orders that impact control room habitability.

3.3 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. See Appendix K for guidance regarding restoration of systems that have been breached.

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APPENDIX M

RESERVED

To be developed

APPENDIX N

GLOSSARY OF TERMS

[APPENDIX DEVELOPMENT IN PROGRESS]

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	?????
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	?????
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?????
DF	Decontamination factor
DNB	Departure from nucleate boiling
DOE	Department of Energy (U.S.)
DSI	?????
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	?????
ESF	Engineered safety feature
FHA	Fuel handling accident
FTC	?????
FSAR	Final safety analysis report

GDC	General design criteria(on)
GL	Generic letter
GSI	Generic safety issue
HELB	High-energy line break
HFP	?????
HPSI	?????
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
JCO	Justification for continued operation?????
LCO	Limiting condition for operation
LEPC	Local emergency planning committee
LOAC	Loss of AC
LOCA	Loss-of-coolant accident
LOOP	Loss of offsite power
LPZ	Low population zone
LRA	Locked Rotor Accident
MHA	Maximum hypothetical accident
MSLB	Main steam line break
MSIV	Main steam isolation valve

MSIVLCS	MSIV leakage control system
MTC	?????
M&TE	Measuring and test equipment
NEI	Nuclear Energy Institute
NHUG	Nuclear HVAC Utilities Issues Group
NIOSH	National Institute for Occupational Safety and Health
NRC	Nuclear Regulatory Commission (U.S.)
NRR	Nuclear Reactor Regulation
OIE	Office of Inspection & Enforcement
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 and 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	?????
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
WASH?????	
X/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 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 forms 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.
- Coreboring.
- Blocking open a door/hatch or damper.
- Modification (addition, removal, or degradation) of a door/hatch or damper.

CRITICAL POWER RATIO: ?????

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: ?????

FILTERED INLEAKAGE: 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 inleakage.

LICENSING BASIS INLEAKAGE: This is the inleakage 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 INLEAKAGE: This is the calculated inleakage 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 INLEAKAGE FOR CONTINUED OPERATION: This is the calculated inleakage 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 must be calculated for each plant in accordance with the requirements of GL 91-18.

MAXIMUM ALLOWABLE TOXIC GAS INLEAKAGE: This is the maximum calculated inleakage of toxic gas that will result in the control room remaining habitable for the bounding toxic gas hazard evaluation.

MAXIMUM HYPOTHETICAL ACCIDENT: ?????

PENETRATION: An opening in a CRE boundary wall or floor/ceiling, other than a door/hatch, which contains materials or mechanical devices which 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: ?????

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: ?????

TRACER GAS (from ASTM E741): A gas that can be mixed with air in very small concentrations in order to study air movement.

UNFILTERED INLEAKAGE: 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 inleakage entering the control room.