UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 7, 2000

MEMORANDUM TO: Cynthia A. Carpenter, Chief
Generic Issues, Environmental, Financial
and Rulemaking Branch
Division of Regulatory Improvements Programs, NRR

FROM: Joseph L. Birmingham, Project Manager
Generic Issues, Environmental, Financial
and Rulemaking Branch
Division of Regulatory Improvements Programs, NRR

SUBJECT: SUMMARY OF JUNE 13-15, 2000, MEETING WITH THE NUCLEAR ENERGY INSTITUTE (NEI) REGARDING CONTROL ROOM HABITABILITY AND NEI 99-03

On June 13-15, 2000, representatives of the Nuclear Energy Institute (NEI) Control Room Habitability Task Force met with representatives of the Nuclear Regulatory Commission (NRC) at the NRC's offices in Rockville, Maryland. Meetings were held in sessions composed of the three Control Room Habitability Task Force subgroups: the Systems Subgroup, the Analysis Subgroup, and the Design Basis Subgroup. Attachment 1 lists attendees for each subgroup.

The Systems Subgroup met June 13. The group reviewed previous work and the status of open action items. They discussed NRC comments on Appendices H, I, J and L, "Toxic Gas Assessments," "Testing Options," "Maintenance Program," and "Vulnerabilities," (Attachments 2, 3, 4, and 5 respectively) and briefly discussed other Appendices. Industry explained they were reorganizing the appendices and that the lettering would change. After discussing the comments with the staff, the comments were resolved or the subgroup agreed to work on wording for future review.

Regarding Appendix J, the staff felt the Appendix discussed inspection of sealing of the control room envelope and did not discuss preventive maintenance of system components. The subgroup stated that discussion on Appendix I, "Testing Options," centered on whether licensees should perform baseline testing or component testing to demonstrate envelope integrity. The staff believes that a baseline test establishes what the overall envelope integrity is. A component test does not. Once the baseline is performed, the staff believes that it may be possible to utilize component testing to demonstrate the status of envelope integrity if it has been established that a correlation exists between the baseline test results and the component testing. On the other hand, the task force believes that a baseline test is unnecessary and that component testing is sufficient as long as the envelope integrity vulnerabilities have been identified. No resolution was reached on this issue during the meeting.

For Appendix H, the subgroup had added a clarification that a new toxic gas assessment is required when a new source of hazardous material is found and a revised toxic gas assessment is required if an inleakage value used for an evaluation is found to be non-conservative. The staff pointed out that no frequency was provided for determining if toxic gas sources in the vicinity of the facility had changed. The staff had previously provided comments to the task force which indicated that the frequency should depend upon plant specific situations. The task force indicated that the frequency would depend on the location of the facility, i.e. rural versus urban, but agreed to review the issue further. Regarding references, the task force chose not to reference proposed draft Regulatory Guide 1087 as it had not been issued for public review. NEI asked if there was a quality assurance pedigree for the use of the EXTRAN component of the HABIT computer code system so that each licensee would not have to develop a pedigree. NRC took an action item for the next meeting to determine if there was a pedigree for HABIT that licensees could use.

On Appendix L, the staff commented that definitions of terms such as inleakage must be consistent throughout the document. The task force agreed and will work with the other subgroups to assure that the definitions are consistent throughout the document.

The Design Basis and Licensing Subgroup met on June 14. After introductions, the subgroup presented a qualitative approach for managing smoke in the control room. The subgroup believes a quantitative approach is inappropriate at this time and presented reasons for their belief. The staff had previously provided comments on the qualitative approach and believes that the approach should be more prescriptive. The task force agreed to review the staff's comments and incorporate them as appropriate. The subgroup will then re-present them to the staff for review. The subgroup then discussed where the regulatory control for the control room habitability program should be; in the technical specifications Part 3, a program described in the technical specifications Part 5 or, part of plant technical reference manuals. The staff discussed placing the program in Section 5 of the technical specifications rather than as specific surveillances with limiting conditions of operations. The staff indicated it would work on a draft technical specification. The subgroup proposed that licensees make a commitment in writing to the NRC to place the program in a licensee controlled program subject to 10 CFR 50.59. No decision was reached on this issue. Attachment 6 contains the subgroup's qualitative approach for control of smoke in the control room with the staff's comments, the subgroup's talking points on smoke control, and the subgroup's proposal regarding regulatory controls for the program.

Regarding review of NEI 99-03, the subgroup indicated that a draft of NEI 99-03 had been provided to the staff before the meeting in May but because the June meeting had resulted in additional revisions industry would forward the next draft to the staff in a couple of weeks. The subgroup asked if future discussions could be conducted by teleconference and the staff agreed that this might be good for the Design Basis and Licensing subgroup but another meeting was probably needed for the Systems Subgroup.

The Analysis Subgroup met June 14 and 15. Attachment 7 contains the proposed agenda; outlines for analyses of containment mixing rates, main steam line break for PWRs, reactor coolant pump shaft lock or break, use of self-contained breathing apparatus for radiological events, control rod drop for BWRs, a maximum hypothetical accident analysis for BWRs, and valve discharges associated with steam generator tube ruptures; a list of potential subgroup accomplishments; and a list of Analysis Subgroup Active Actions. Attachment 8 is two NRC slides on fuel gap fractions discussed during the meeting. The subgroup also discussed the criteria for analysis of a fuel handling accident and the criteria for determining the duration of an iodine spike accident. Although additional evaluation of the criteria for the scenarios presented needs to be performed, the NRC found the outlined analyses to be a good start.

During the meeting on June 15, the subgroup again discussed placement of regulatory controls for control room habitability programs. No new positions were proposed and no agreement for placement of the controls was reached.

New business discussed by the subgroup included use of ARCON 96, nonconservatisms, the quality of meteorological data as the physical environment of a site changed, and a question on the use of the center point versus the exterior of the containment as a dose source. The subgroup then discussed topics and format for future meetings and the meeting was adjourned.

Project No. 689

Attachments: As stated

cc w/atts: See list

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List of Systems Subgroup Attendees
June 13, 2000

NAME	ORGANIZATION
John Hayes	NRR/DSSA/SPSB
Hal Walker	NRR/DSSA/SPSB
Joseph Birmingham	NRR/DRIP/RGEB
Steve LaVie	NRR/DSSA/SPSB
Paul Boehnert	NRC/ACRS
Steve Thomas	Northern States Power
Jerry Kloecker	Virginia Power
Jim Riley	Nuclear Energy Institute
Kris Parczewski	NRR/DE/EMCB
Bernie Jwaszewski*	Vermont Yankee
John Wynn	Baltimore Gas & Electric
Brad Harvey	Duke Energy

List of Analysis Subgroup Attendees
June 14 and 15, 2000

NAME	ORGANIZATION
Syed Ahmed	Dominion Generation
Kurt Cozens	Nuclear Energy Institute
Ken Jha	Bechtel
Tom Mscisz**	PECO Nuclear
Jim Metcalf	Polestar
Jerry Gryczkowski	Baltimore Gas & Electric
Steven P. Schultz	Duke Energy
John W. Cotton	Entergy
R. Brad Harvey*	Duke Engineering Services
Mark Reinhart**	NRC/DSSA/SPSB
Steve LaVie	NRC/DSSA/SPSB
Michelle Hart	NRC/DSSA/SPSB
Leta Brown	NRC/DSSA/SPSB
Mark Blumberg	NRC/DSSA/SPSB
Jack Hayes	NRC/DSSA/SPSB
Alexander Dromerick	NRC/DSSA/SPSB
Joe Birmingham	NRC/DRIP/RGEB

* June 14 only
** June 15 only

List of Design Basis & Licensing Subgroup Attendees
June 14, 2000

NAME	ORGANIZATION
Mark Reinhart	NRC/NRR/SPSB
Jack Hayes	NRC/NRR/SPSB
Mark Blumberg	NRC/NRR/SPSB
Steve LaVie	NRC/NRR/SPSB
Harold Walker	NRC/NRR/SPLB
Mark Henry Salley	NRC/NRR/SPLB
Joe Birmingham	NRC/NRR/RGEB
Naeem Iqbal	NRC/NRR/SPSB
Paul Boehnert	ACRS Staff
Jim Riley	NEI
Jerry G. Sims	Southern Company
John Duffy	PSEG Nuclear
Mike Ruby	RG&E
Kurt Cozens	Nuclear Energy Institute
David J. Distel	PECON
Jerry Burford	Entergy
Duane Gore	STP/Stars Group

APPENDIX H

Toxic Gas Assessments

1. Purpose

This appendix provides guidance for utilities in assessing the habitability of the control room during and after a postulated external release of hazardous chemicals. *A new toxic gas assessment is required if a new source of hazardous chemicals is found in the vicinity of the plant; a revised toxic gas assessment is necessary if an inleakage value used in an existing toxic gas evaluation is found to be non-conservative.*

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 document.

3. Regulatory Basis

Criterion 4, "Environmental and Missile Design Basis,"of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, 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. Definitions

5. 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 ~~Draft~~ Regulatory Guide ~~DG-1087 (Reference 6.1). This Draft Regulatory Guide combines two existing regulatory guides,~~ 1.78 (Reference 6.2)~~ and 1.95 (Reference 6.3), since much of the language on regulatory positions is common in these two guides. Draft Regulatory Guide DG-1087 also revises Regulatory Guide 1.78 in several areas, including specification of toxicity limits and recommended use of an updated atmospheric dispersion model.~~ This appendix also provides

additional guidance beyond that contained in ~~Draft~~ Regulatory Guide ~~DG-1087~~*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 models, including* dense gas atmospheric dispersion models. *Licensees have the choice of either using the methodology that currently serves as their licensing basis, the guidance presented in this appendix, or any other regulatory guidance that may subsequently be published by the NRC.*

5.1 Identifying Hazardous Materials

5.1.1 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.

Internet sources of data on hazardous materials 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

5.1.2 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.

5.1.3 Toxic Limits

The hazardous chemical toxicity limits that ~~should~~*can* be used for control room evaluations *are either those listed in Table C-1 of Regulatory Guide 1.78* ~~are~~*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.[1] For the purposes of conducting control room habitability evaluations, the IDLH limits ~~are~~*should be* implemented as 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).

5.2 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 ~~Draft~~ Regulatory Guide ~~DG-1087~~*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.

[1] The purpose of establishing IDLH exposure concentrations was to ensure that workers can escape from a given contaminated environment in the event of a failure of respiratory protection equipment.

5.2.1 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 ~~Draft~~ Regulatory Guide ~~DG-1087~~*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.[2] A detailed evaluation should be performed for those hazardous chemicals stored in weights greater than 100 pounds either onsite or within 0.3 miles of the control.

- *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 of hazardous chemicals within a five-mile radius of the plant in quantities exceeding those shown in Table C-2 of ~~Draft~~ Regulatory Guide ~~DG-1087~~*1.78* (adjusted for the appropriate toxicity limit, meteorology, and control room air exchange rate) should be considered in a detailed evaluation of control room habitability. 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 calculations 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.

5.2.2 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 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*~~is~~ 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).

[2] Appendix A to ~~Draft~~ Regulatory Guide ~~DG-1087~~*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.

If detailed calculations show that ~~IDLH exposure~~*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.[3] 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.[4] The detail 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 ~~IDLH exposure~~*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 ~~IDLH concentration~~*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 and located at a nearby stationary facility; 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 and frequently transported near the site; 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.

- *Release Characterization.* The release classification 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 and pre-released storage conditions. 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

[3] 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 ~~IDLH exposure~~*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).

[4] The American Industrial Hygiene Association (AIHA) has established odor thresholds for a number of toxic chemicals (Reference 6.7).

mixture. Conversely, chemicals stored as a gas may partially or completely condense to form liquid droplets when released. The degree of condensation, if it occurs, will again depend on the pre-release conditions and physical properties of the material. 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.

Appendix B to Regulatory Guide 1.78 presents algorithms for performing atmospheric dispersion modeling for the detailed evaluation of control room habitability for potential toxic gas releases. As an alternative, ~~Draft Regulatory Guide DG-1087 suggests the use of~~ the EXTRAN atmospheric dispersion model contained in the HABIT computer code system (References 6.8 and 6.9) *can also be used.* ~~for the detail evaluation of control room habitability for potential toxic gas releases.~~ An advantages of utilizing the HABIT computer code system is that the output from EXTRAN is linked directly to another HABIT computer system code, CHEM, which calculates the resulting chemical exposures to the control room personnel.

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.*

- ~~Draft~~ Regulatory Guide ~~DG-1087~~*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 6.10), "Risk Management Program Guidance for Offsite Consequence Analyses" (Reference 6.11), and "Guidance on the Application of Refined Dispersion Models to Hazardous/Toxic Air Pollutant Releases" (Reference 6.12). The later reference also provides guidance on how to execute several generally available dense gas atmospheric dispersion models.

5.3 Toxic Gas Protection

- Toxic gas detectors
- Toxic gas ventialtion system operational mode
  - Emergency zone isolation
  - Bottled air supply
  - Air purification systems
- Individual respiratory devices
- Time to don respiratory protection
- Protective clothing

6. References

6.1 Draft Regulatory Guide DG-1087, "Revision 1 to Regulatory Guide 1.78, Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," U.S. Nuclear Regulatory Commission, June 1999.

6.2 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.

6.3 Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," U.S. Nuclear Regulatory Commission, January 1977.

6.4 H.R. Ludwig, S.G. Cairelli, and J.J. Whalen, "Documentation for Immediately Dangersous to Life or Health Concentrations (IDLH)," National Institute for Occupational Safety and Health, 1994.

6.5 National Institute for Occupational Safety and Health, "NIOSH Pocket Guide to Chemical Hazards," U.S. Government Printing Office, 1997.

6.6 NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission, July 1981.

6.7 American Industrial Hygiene Association (AIHA), "Odor Thresholds for Chemicals with Established Occupational Health Standards," AIHA,1989.

6.8 SA Stage, "Computer Codes for Evaluation of Control Room Habitability (HABIT)," NUREG/CR-6210, Pacific Northwest National Laboratory, June 1996.

6.9 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.

6.10 EPA-454/R-92-024, "Workbook of Screening Techniques for Assessing Impacts of Toxic Air Pollutants (Revised)," U.S. Environmental Protection Agency, December 1992.

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

6.12 EPA-454/R-93-002, "Guidance on the Application of Refined Dispersion Models to Hazardous/Toxic Air Pollutant Releases," April 1993.

Action 22
Appendix I
Testing Options

1. Purpose

This Appendix describes testing options available to licensees to demonstrate that the control room habitability system conforms to the plant licensing/design basis.

2. 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.

3. Definitions

For the purpose of this guidance document, several terms will be used to describe inleakage as follows:

a. Licensing Basis Inleakage: This is the in-leakage that is used in the plant design basis radiological analysis with design basis values of other plant parameters to calculate control room operator dose during a licensing basis accident.

b. Maximum Allowable Radiation In-leakage: 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.

c. Maximum Allowable Radiation In-leakage for Continued Operation: This is the calculated in-leakage value in cfm that will result in the control room operators receiving the maximum allowable dose with realistic but verifiable inputs of all other parameters to the plant radiological analysis. This value must be calculated for each plant in accordance with the requirements of GL 91-18.

c. 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.

d. 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. An example would be air that leaks (infiltrates) as control room doors are opened and closed. (note that in a control room configuration without vestibule doors or an airlock; the pressurized control room will lose ALL pressure when a door is opened. When this happens the pressurized control room momentarily is no longer capable of preventing unfiltered inleakage. Therefore, unfiltered inleakage will occur.) Radionuclides are not removed from this air prior to it entering the control room.

d. 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.

4. System Mode of Operation

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. 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.

Two common modes of operation are pressurization and isolation. 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.

Actual in-leakage should be determined in each system mode of operation and compared to the in-leakage for the particular event being evaluated. For radiological events, the actual in-leakage should be compared to the licensing basis in-leakage and possibly to the maximum allowable in-leakage, depending on the test results. For toxic gas events, the actual in-leakage should be compared to the allowable toxic gas in-leakage.

5. Test Methods

a. Tracer Gas Testing

One test method is tracer gas testing in accordance with ASTM Standard E741-95. The title of this standard is "Standard Test Method for Determining Air Change Rate in a Single Zone by Means of a Tracer Gas Dilution."

Reference Appendix M for additional information concerning tracer gas testing.

b. Alternate Test Methods

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

- Test potential leak paths and produce an overall in-leakage value in cfm for the entire system envelope.
- Performed in accordance with an established test standard.
- Performed with systems operating in the post-accident configuration.

The proposal of alternate test methods is necessarily plant specific due to the many variations in habitability system design and must be established by each licensee.

Licensees that propose to demonstrate conformance with the regulations using methods other than tracer gas testing in accordance with ASTM Standard E741-95 should include a thorough description of the proposed methods and detailed technical justification for the proposed method.

See the attached table 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.

6. Test / Assessment Frequency

Licensees are required to verify the adequacy of design by 10 CFR 50 Appendix B. A baseline test to quantify leakage into the control room habitability system will satisfy this verification requirement. The need for periodic, follow-up tests or assessments should be established based on initial results, vulnerabilities, repeatability of results, system modifications, envelope changes, maintenance practice changes and margin. If a large margin exists between the maximum allowable radiation in-leakage and the licensing basis radiation in-leakage, a longer period between tests or assessments is justified. Conversely, if the margin is small, more frequent testing may be needed.

All plants should have programs to manage the effects of aging of components. These programs can be tests, inspections or combinations of tests and inspections. Plant vulnerabilities should be considered with maintenance programs to determine the content of these aging management programs and the appropriate period of time between follow up testing and/or assessment.

For example, a plant with actual leakage of 40 cfm, a licensing basis inleakage of 500 cfm and a habitability system that has all equipment inside the pressure boundary with only welded ductwork outside the pressure boundary should have a relatively long period between tests or assessments.

Conversely, a plant with actual leakage of 250 cfm with a licensing basis inleakage of 300 cfm and a habitability system that has air handlers, fans and non-welded ductwork outside the pressure boundary, should test or assess more frequently.

Appendix I
Table 1
Available Test Methods

Type of Test	Standard	Comprehensive	Advantages	Disadvantages	Performed under limiting condition	Accuracy	Quantitative	Comments
Tracer Gas (SF6)	E741	See Comments	History NRC Prefers	1. Wind Effects 2. Disrupts Plant 3. High Cost 4. Leak location not identified 5. Tests from inside out	Depends on plant	±10%	Yes	1.
Pressure Test (Blower Door)	E779	?	1. Performed under pos. or neg. press. 2. Requires CRHVAC shutdown. 3.	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	E1186	No	1. Identifies location 2. Inexpensive 3. No effect on OPS		Yes.	N/A	No.	
Component Test	E779 E1186 E741	No	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.

APPENDIX J

CRE Maintenance Program Guidance

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. First, the amount of inleakage (filtered and unfiltered) may significantly affect the post-accident radiological dose to the operators. The safety analyses assume a certain value for inleakage during and following a Design Basis Accident to demonstrate that the operator dose remains within allowable limits (typically GDC 19). The CRE and associated mechanical components must be able to maintain the inleakage so as to meet these limits. The CRE must also provide protection for the operators against a smoke or toxic gas event. Second, 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 N, will provide guidance in this area. For example, if a CRE is pressurized following a DBA and an adjacent cable spreading room is at a lower pressure than the CRE, minor leakage through cable penetrations to the room will be out 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 into the CRE. Depending on the presence and location of charcoal filters, this inleakage may be unfiltered, and could significantly raise operator dose.

2.1. CRE BARRIER CONTROL

Control of the CRE pressure boundary should be maintained at all times. 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. The procedure should include the total allowable open areas such as open door or wall penetration areas, as well as temporary measures required prior to breach of the pressure boundary. The required operation or status of ventilation components, including non-safety equipment whose normal operation affects the boundary, should be noted. Upon completion of breach activities, a post-modification

test of the boundary (e.g., verification of normal parameters such as supply fan flow or CR-outside pressure differentials) is highly recommended.

2.1. SEALING PROGRAM DEVELOPMENT

A CRE assessment, as outlined in Appendix N, should be performed to assess the vulnerability of the envelope to leakage, and identify penetrations by type and expected sealing function during postulated accidents. The assessment should include a review of applicable building and system drawings, walkdowns, and discussions with Maintenance, Operations, and Engineering personnel. This information can then be used to identify all penetrations, as far as practical, 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 following is a list of typical penetrations. Note that leakage through the penetrations may be either in or out of the envelope, depending on the CR design and operation of other equipment (possibly NS) during an accident.

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

The method and frequency of inspection/repair/modification will depend on the type and safety significance of the penetration. Basic guidelines 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.

2.2. DOORS AND DOOR SEALS

The door must fit properly in the frame, with hinges securely attached. Door sweep must be in continuous contact with the floor or threshold for the entire width of the door. The gasket or seal must be an approved type, be free of cracks, and must form a contact seal around the entire perimeter of the door. The door and frame must be free of breaks or open holes. With the door closed, the seal must be compressed against the door at all points.

2.3. DAMPERS

Dampers, associated linkages, and actuators shall 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 shall 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, must be an approved type, be free of cracks, and must form a contact seal around the entire perimeter of the damper or where installed. The damper and frame must be free of breaks or open holes. With the damper closed, the seal must be evenly compressed against the damper at all points.

2.4. 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 openings are frequently difficult to find. Repairs should be made using approved sealants or grouts, in accordance with vendor instructions.

2.5. 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 ductwork or fan. Clamps and ductwork/fan connections should be inspected for defects that

could adversely affect proper sealing. 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 must be rated for sun and weather exposure, or be covered with an approved coating.

2.6. 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.

2.7. 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.

2.8. AHU / FAN HOUSINGS AND SHAFTS

Inlet and outlet flanges should be sealed with approved sealants, or preferably continuously welded on both sides. Any fan housing drains should have plugs installed. AHU drain loop seals should be verified periodically. Separate sections of AHU housings should have individual drains. High quality or double gaskets (not sealants) should be used on cover plates and access doors. Bolts on cover plates and access doors should be spaced on 3" to 4" centers. Recommended shaft seals are stuffing box seals, lip seals, or mechanical type seals. An arrangement using a neutral purge gas is also effective.

3. REFERENCES

1) ANSI N510 Testing of Nuclear Air Cleaning Systems

2) ASME AG-1 Code on Nuclear Air and Gas Treatment

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

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

5) NHUG Draft Guidance on Breach Control

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
VULNERABILITY ASSESSMENT

Purpose:

The purpose of this Appendix is to provide guidance for identifying potential vulnerabilities to inleakage.

Scope:

This appendix only provides guidelines to assist personnel in identifying potential vulnerabilities to inleakage into the Control Room envelope. These potential vulnerabilities have been identified through testing, experience at several plants, and personnel knowledge. This appendix does not provide guidance for minimizing these vulnerabilities. Appendix J provides the guidance for sealing or minimizing these vulnerabilities.

Regulatory Basis:

Not sure what to put here?

Definitions:

For consistency within the entire document, the definitions should be included in the main body of the Document. The following terms used in this Appendix should be included in the list of definitions:

Inleakage:

Control Room Envelope:

Pressurized Control Room (or Positive Pressure Control Room):

Neutral Pressure Control Room:

Note that the definitions for pressurized vs. neutral pressure control room should be based not on the actual pressures in the rooms, but on the system configurations. That is, a pressurized control room is equipped with a pressurization system while a neutral control room is not. Using this differentiation may help to dispel some of the perceived confusion.

~~Body of Appendix~~Identifying Vulnerabilities

Table L-1 provides a list of items to consider when determining potential vulnerability susceptibility. This listing is provided to include identified potential system and envelope configurations. Depending on system and envelope configuration not all of these may be

applicable to any given facility. It is possible that not all potential vulnerabilities have been identified in the table or the following text. The user of this information should be careful not to use this as complete listing for their facility, but apply it as guidance for the types of vulnerabilities to look for.

The additional description below is to aid the user in the use of the table; specifically, where the item may not be clear by itself.

1. Control Room Pressure

Fluids (in this case air) flow from areas of high pressure to areas of lower 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 when identifying the potential sources of inleakage. This is valid regardless of the ventilation system design (pressurized control room or neutral pressure control room).

For a positive pressure envelope design, leakage through the envelope boundaries are from the envelope to the adjacent areas (provided the space in the envelope is at a higher pressure relative to the adjacent space). A positive pressure control room does not ensure that inleakage cannot occur, it just indicates that an air source outside of the envelope is entering the envelope. Inleakage can occur at ventilation ducting, dampers, drain lines, etc, or from other systems that traverse the envelope. Note that excessive outleakage from the envelope is also an important consideration as this places an increased demand on the pressurization system and increases the filtered inleakage value in the dose analysis.

For a neutral pressure envelope design, leakage through the envelope boundaries can be either in or out, depending on the direction of the differential pressure. Inleakage is obviously a concern. Excessive outleakage from the envelope is also an important consideration as this must be off set by inleakage through other boundaries.

To determine if there are any adjacent areas which could be at a higher pressure than the rooms within the envelope (note that elevation changes may also affect pressure differential and need to be accounted for ~~along with temperature differences~~) measure the differential pressure between the envelope and the adjacent spaces. Items to consider when measuring the differential pressure:

- The preferable method to measure is 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.
- Measure the pressures in all adjacent areas to the envelope.
- Measure the pressure in all rooms within the envelope. Take enough measurements within a given room to ensure that there is not large

variations in the room and that the room is actually positive. For example, complicated room configurations with restrictions to air flow (panels, half walls, etc.) can result in pressure variations within the envelope, return header drawing air from a room within the envelope can result in a negative pressure in that room,

During review of the pressures in the envelope and adjacent areas, consider all possible system configurations (of the control room ventilation system and the ventilation systems in adjacent areas). Items which have been overlooked previously which should be factored into this review include:

- Determining automatic and/or manual response of the system to different events (safety injection signal, high radiation signal, toxic gas, etc.). For example,

  a) A control room envelope could be pressurized during a radiological event and not pressurized during a toxic gas event.
  b) 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.
  c) 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).

- System alignments accounting for single active failures. 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 inleakage perspective. For example,

  a) For a neutral pressure control room, running both trains can result in an increased number of rooms which have negative pressure relative to the adjacent areas.
  b) For a positive pressure control room, running both pressurization systems can result in increased unfiltered inleakage if the fans are located outside the envelope and shafts are not sealed.

- Ventilation system alignments in adjacent compartments should consider the most limiting configurations. These should account for cases with and without off site power. Consideration of a loss of off site power (LOOP) coincident with the event should be consistent with the licensing basis for the facility. ~~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. That is, during different seasons (hot or~~

~~cold outside), the ventilation systems serving these areas may be operated in different configurations.~~ For example:

a) Assuming a LOOP may result in more favorable system alignments in some ventilation systems for adjacent spaces and those routed through the envelope.

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 (hot or cold outside) or day-time vs. night-time , the ventilation systems serving these areas may be operated in different configurations.

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 J(?).

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).

2. Control Room Ventilation System

Inleakage can occur into portions of the ventilation system which are located outside of the envelope; specifically, if portions of these systems (e.g., return ducting) are at a negative pressure relative to the area(s) it is 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.

AHU housings can be a source of inleakage if it is not welded or the integrity is compromised. Specifically inspect the underside as this can be a location of corrosion due to moisture accumulation.

AHU door seals, electrical and instrumentation penetrations can be a source of unfiltered inleakage.

AHU and ventilation system doors, hatches, etc. can be a source of unfiltered inleakage. Inspect such items as latches, sealing surfaces, seal compression, etc.

Fan shafts which penetrate the housing can be a source of inleakage.

Ducting that is isolated can be source of unfiltered inleakage is the isolation dampers are not leak tight. discuss in relation to fan pressure

3. 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 inleakage if the systems 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 operating mode of the subject ventilation system to isolate the ducting during an event or ensure that it operates with a lower pressure than the envelope pressure.

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.

4. Control Room Envelope Boundary Penetrations

Penetrations such as cables and conduits, small pipes, etc. can be a potential source of inleakage. To the extent practical, both the inside of the conduit and the conduit/wall penetration should be inspected.

Concrete anchors through block walls may not be sealed and 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(s) is(are) appropriate for this application.

5. Doors in Control Room Envelope Boundary

Door seals can be a potential significant source of inleakage. 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.

6. Ventilation System Isolation Dampers

Control Room Ventilation System isolation dampers which close to ensure the integrity of the system and the envelope during an event can be potential sources of inleakage. Redundant dampers should exist at each location to meet single active failure criteria.

Louvered dampers have historically demonstrated to be unreliable isolation devices. This does not mean that these types of dampers are unacceptable. It is listed here to note that louver dampers are more susceptible to leakage. This also does not imply that other types of isolation dampers cannot be a source of leakage.

Leakage can also occur through damper shafts.

7. Other Systems in the Envelope

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

8. General Boundary Construction

Certain construction configurations or deficiencies are more susceptible to inleakage. For example, porous (non filled) block walls can leak where poured concrete will not leak. Deficiencies such as cracks or inadequate sealing materials can be locations for inleakage. 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 looked at to the extent practical, and it may also be possible to verify the boundary by looking at the other side.

~~9. Boundary Control Program~~

~~The absence of an effective control room envelope boundary control program can inadvertently allow open penetrations, or poorly sealed penetrations, in the envelope boundary. Controls need to be in place to identify and track breaches to the envelope boundary. These controls should include verification that the breach was closed and the boundary returned to an acceptable condition. The boundary control program also ensures that (as appropriate) Technical Specifications, compensatory actions, administrative controls, etc. are adhered to while the breach is present.~~

TABLE L-1 ~~(?)~~
DETERMINATION OF VULNERABILITY SUSCEPTIBILITY

Vulnerability	Determining Vulnerability Significance
Control Room Ventilation System Integrity	Determine if any control room ducting and/or HVAC equipment located outside the envelope is at a negative pressure with respect to adjacent areas. Determine if AHU fans which continue to operate during an emergency have the potential to draw air from isolated ducting lines (i.e., damper leakage) which penetrate the envelope boundary. Determine if a potential exists for isolated ducting to serve as a conveyance vehicle for unfiltered in-leakage due to natural circulation. ~~Verify integrity of~~ Inspect ~~-~~ducting and equipment located (e.g., AHU, recirculation fan) outside of the envelope for potential vulnerabilities. Determine if any ventilation system penetrations or doors, hatches, etc. could be potential sources of unfiltered inleakage. Determine if AHU fan shaft inleakage could result in unfiltered inleakage. Determine if recirculation fan shaft inleakage exists and could invalidate filter flow requirements in the emergency mode of operation. Determine if any pressurization ducting between the envelope boundary and the filter is operated at a higher pressure than the envelope pressure (this is applicable if the ductwork is located within the boundary).
Control Room Ventilation System Operation	Determine if any ventilation systems are improperly balanced and could provide undesirable differential pressures across the envelope boundary.
Other Ventilation Systems	Determine if there is any ducting routed through the envelope ~~that is not isolated~~ 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 being a means of in-leakage or out-leakage) Determine if any of this ducting is not seam welded or has welding deficiencies
Penetrations in the Envelope Boundary	Determine if there are any wall (including floor and ceiling) penetrations (i.e., conduits, electrical cable trays, etc.) that are unsealed and are potential sources of leakage. Check seals

	inside the conduit and between the conduit and the wall. Determine if there any openings around ventilation ducting penetrations and dampers that are not properly sealed. Determine if there are any drains (floor or equipment) which lack loop seals or check valves and can provide a leakage pathway. If used, verify that the check valve(s) is(are) appropriate for this application. Determine if there are any other types of penetrations which can provide potential leakage pathways. For example, concrete anchors through block walls which are not sealed,
Envelope Doors	Determine if there are any defects in any doors. Determine if any door seals are cracked, are missing seals or have loose or improper fitting seals. Determine if any door is not properly compressed or fitting against the door seals. Determine if any door latches are not functioning properly to maintain the door securely closed. Determine if any door frame is not properly sealed.
Isolation Dampers	~~Determine if there are any penetrations into the envelope with only single isolation dampers or louvered dampers.~~ Determine if any control room isolation damper seals are cracked, are missing seals or have loose or improper fitting seals. Determine if any control room isolation damper linkages are not functioning properly to assure compression of the seals against the damper blade(s). Determine if damper shafts can be a potential source of unfiltered inleakage.
Other Systems in the Envelope	Determine if there are any instrument or service air lines which enter the envelope boundary and could provide potential unfiltered inleakage sources due to leakage, operational venting of air operated components, etc.
General Boundary Construction	Determine if the envelope has any boundary or barrier discrepancies (check concrete, block, expansion joints, etc.)
~~Boundary Control Program~~	~~Determine if a program is in place to control envelope breach time, opening size, compensatory actions and post-work inspections ensures envelope integrity is not compromised during maintenance.~~ ~~Determine if a program is in place to perform periodic inspection of the assurance of boundary~~

SMOKE

- NEI 99-03 will address smoke as a generic issue and provide qualitative guidance assessing smoke infiltration vulnerability. Included in this approach will be the acknowledgement of the prudence of reducing inleakage because of the potential for smoke infiltration.

- Evaluate and incorporate appropriate NRC comments on the draft guidance

- The task force believes it is inappropriate to develop quantitative guidance at this time because:

    - Beyond the scope of the task force
    - The following criteria do not yet exist

        - Type of fire/smoke
        - Location
        - Duration

TECHNICAL SPECIFICATION

- We have read your response. We were expecting some discussion on different control mechanisms for inleakage, which is a design input. The comments presented a TS SR as a control mechanism without stating why other mechanisms are not acceptable.

- The Task Force has reviewed this extensively:
  - Spoke with the TSTF and others
  - Inleakage is not a parameter that meets 10 CFR 50.36 criteria

- NEI 99-03 when issued will set expectations for a licensee to commit to establish a program to test, maintain, and periodically assess the CR envelope:

  - baseline testing to quantify unfiltered inleakage and inleakage
  - periodic reassessment of CR envelope integrity

- The TF expects that once the industry responds to NEI 99-03 the integrity of the CR envelopes will be improved.

- The TF believes that a licensee program will meet all the 10 points in the NRC write-up with the exception of 4 and 7

DRAFT (4/25/00)

In-leakage of Externally Generated Smoke
Qualitative Approach (NEI 99-03)

As written, anyone reading this white paper would conclude that the inleakage of externally generated smoke to the control room is an issue which need not be addressed and one would even question why there is a white paper. As presently written, the white paper defines the problem and then provides a decision outline in the case of an event. However, the decision outline is incomplete and will be discussed later in these comments. In addition, there is an essential element missing from the white paper. That element is the assessment which determines the plant-specific vulnerabilities to the problem. That assessment, is critical so that the licensee can determine which vulnerabilities exist and so the licensee can implement solutions to those vulnerabilities. Therefore, the staff suggests that the white paper be structured to include a definition of the problem; guidance to assess potential vulnerabilities to the problem; and guidance on a program management plan to address these vulnerabilities.

With the amount of measured in-leakage greater than estimated at a number of plants, the NRC developed a concern that there may be an increased risk of smoke infiltration from an external source. The smoke is postulated to be from a source such as a transformer or turbine lube oil fire. Further, the concern is that in some cases, the smoke could make the Control Room uninhabitable and affect or block the Operator's egress to the Remote Shutdown locations. The Task Force (TF) believes that this is an extremely unlikely event and does not believe that it rises to the level of a generic issue, but does acknowledge that the consequences of such an event could be serious if not properly mitigated. Further, the TF believes that it is not feasible to quantify an allowable concentration of smoke and then arrive at an engineered solution. With this in mind, the facility should ensure that there is a success path for dealing with such an event in a qualitative manner. It is not anticipated or intended that this issue result in a design change beyond that which has already been accomplished to comply with Fire Safe Shutdown (Appendix R) requirements unless there is a compelling reason to do so.

Suggest the following for consideration. Initial paragraph seems to jump into the midst of the issue too quickly and there is defensiveness -in its presentation. Presentation says it is a highly unlikely event, it is not generic, not feasible to quantify the allowable concentration of smoke, and concludes that it is unlikely that this potential situation will require a design change. The staff would suggest a more gradual introduction into the problem starting with the regulatory requirements and then defining how success is achieved. For example, one might say, "One of the requirements of GDC 19 is that the control room operators have the capability to control the reactor from either the control room or the remote shutdown areas. Success is measured by having either or both locations available to perform such a task". Then the next step might be to identify examples of the problem. It seems like there are at least two of which we aware. The first is the identification of the lack of control room envelope integrity. This has raised concerns as to whether the control room would be

available in the event of a fire in either the Aux or Turbine Bldgs and in areas where remote shutdown functions would be performed. The second is that a recent inspection at the St. Lucie Plant identified a situation where a fire in the cable spreading room, which is located within the control room envelope at St. Lucie, could have negated the capability of the remote shutdown functions. This was due to 1) the fact that the shutdown panel was located adjacent to the cable spreading room 2) the cable spreading room door being a pathway for controlling the fire; and 3)the inadequacy of the fire barriers between the cable spreading room and the remote shutdown panel room. A third might be the migration of fire suppressants into the control room following their activation.

With respect to regulatory success, it would seem that the guidance to licensees would be to assure that one of these shutdown control pathways continues to exist and that there is a minimization of reliance on SCBAs. No fire should be capable of removing both shutdown areas from operation.

With respect to some of the other statements made in the White Paper's initial paragraph, the following comments are provided. The concern is not limited to the control room being uninhabitable and egress to the remote shutdown locations being blocked or affected. The concern is that neither the remote shutdown nor the control room are available for shutdown. Whether the described scenario is a low probability event is immaterial. A LOCA is a low probability event yet reactors are designed for that event. And a fire is not a low probability event. It is an anticipated operational occurrence and the number of fires have been significant. However, it is true that the number of significant fires has not been numerous.

This white paper, which is anticipated to become an Appendix, should provide information about an anticipated operational occurrence which may not have been adequately addressed because one component of the information dependent upon to resolve the issue (control room envelope integrity) was faulty, i.e., the control room envelope integrity is less than assumed. Since all of the plants which have tested their control room envelope integrity have exceeded their assumed value for inleakage, this particular issue would seem to warrant consideration by all licensees. One could state that it is likely that Appendix R did not address this issue mostly because it was assumed that the control room envelope had integrity.

Before dismissing the feasibility of quantifying an allowable concentration of smoke, it would seem appropriate to first identify the challenges, identify possible solutions, and then ascertain whether it is necessary or appropriate to perform calculations to implement the solution. It is premature to presume that licensees will not have to implement any design changes because it is not apparent that licensees have addressed the issue of assuring that reactor shutdown capability continues to exist even when control room envelope integrity is absent . Without data from such assessments, it would seem to be difficult to draw any conclusion.

The steps below outline some possible success paths to safely shutdown the plant during a smoke

infiltration event. A flow chart of the process is contained in Attachment 1.

As noted in previous comments, prior to this section there needs to be a section which describes a program which assesses vulnerabilities. Examples should be given of the problems and potential solutions, if known from those licensees which have addressed this issue.

As structured, Attachment 1 does not show the appropriate success paths. They are that either the main control room or the remote shutdown facilities are available to shutdown the reactor. As presently structured, success is measured by being able to get to the remote shutdown areas. You may be able to get there but you may be unable to perform your duties once there.

1. Should an excessive amount of smoke infiltrate the Control Room Envelope, the Operators would isolate the system and assess the habitability. They would face a decision of whether to stay in the Control Room and attempt to clear the smoke or 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 clean source of purge air.

   Seems like an assessment would need to be made as to whether to isolate or to purge the control room rather than to automatically assume to isolate the control room. It should be noted that the isolation of the control room ventilation system may not result in the elimination of the smoke source. In a fire situation, time is of the essence.

   The decision to abandon the control room should be based upon visibility in the control room and the conditions at the remote shutdown panels. It also seems that this decision process should be immediately initiated with the intrusion of smoke. It may also be necessary for the intrusion of CO and fire suppressants. Guidance which states that the operators need to start taking actions when the concentration of smoke becomes excessive is too subjective. How does one define excessive and is it the same for each facility?

2. If the operators decided to stay in the control room, it would be necessary to purge the smoke from the envelope. This could be accomplished by the ventilation system in the purge mode. If all Control room fresh air intakes are contaminated with smoke, it may be possible to open a door and evacuate the smoke using portable smoke movers and "elephant trunks". A short-term limited use of SCBA's may be expected in this situation. Clearing of the smoke in a reasonable period of time would be considered a success path.

   Depending upon the plant design, the purge mode of operation may not exist at a facility. One should not presume that because one has certain equipment and potential solutions that their facility is immune to the smoke infiltration problem. As an example, one may have smoke ejectors. They can be utilized to remove smoke. However, if utilized, does their utilization result in fire barriers being breached? Where does the smoke go when the ejectors are utilized? Can they handle the quantity?

How long does it take to remove the smoke? This is just one example of how a complete assessment is required.

3. If attempts to clear the control room atmosphere are unsuccessful, a re-evaluation of the evacuation option may be in order. 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.

   It would appear that there are certain issues which need to be addressed. For example, is there damage in the remote shutdown areas? What changes are occurring relative to the generation of smoke? Is the amount of smoke increasing or decreasing? Does it contain toxic chemicals? Does it contain CO? It may be necessary to assess the damage in the areas where there was smoke. The possibility exists that these areas contained fire damage.

4. If the operators decide to evacuate the control room, a path must be clear to the remote shutdown station(s).

5. An alternate route should be pre-identified in case the source of the smoke is also affecting the normal evacuation path. It is not intended that the backup route be equipped with emergency lighting to specifically cover this scenario.

6. In the event a backup route is not available, it will be necessary to transit the smoke area using SCBAs.

7. To ensure success in manning the Remote Shutdown Panels, it must be assumed that the remote stations remain habitable. Although SCBAs may be relied upon for a limited period, equipment should be available to clear the atmosphere in these locations in a reasonable period of time.

   Besides visibility, should this item address other concerns? For example, does temperature need to be addressed?

The Facility should review the following to ensure the required actions can be successfully completed:

- Verify that sufficient procedural guidance exists, with the appropriate level of detail, for the Operators to perform the necessary tasks.

- Verify that initial and continuing training is performed to ensure familiarity with the processes involved. The training should include instruction on procedural guidance and equipment usage.

  These items should include the assessment of potential challenges and the alternate pathways to the remote shutdown areas. Realistic drills should be performed which

include cylinder usage and changeout, vision impairment, SCBAs, communication problems and the use of alternate pathways to the remote shutdown areas.

- Certain success paths contain situations that require the limited use of SCBAs. Verify that equipment is available and that a sufficient number of Control Room Operators are qualified in their use to safely shutdown the plant.

  It should not be a "sufficient number" but rather all control room operators should be qualified to safely shutdown the plant.

- Verify that the appropriate equipment is available and properly staged.

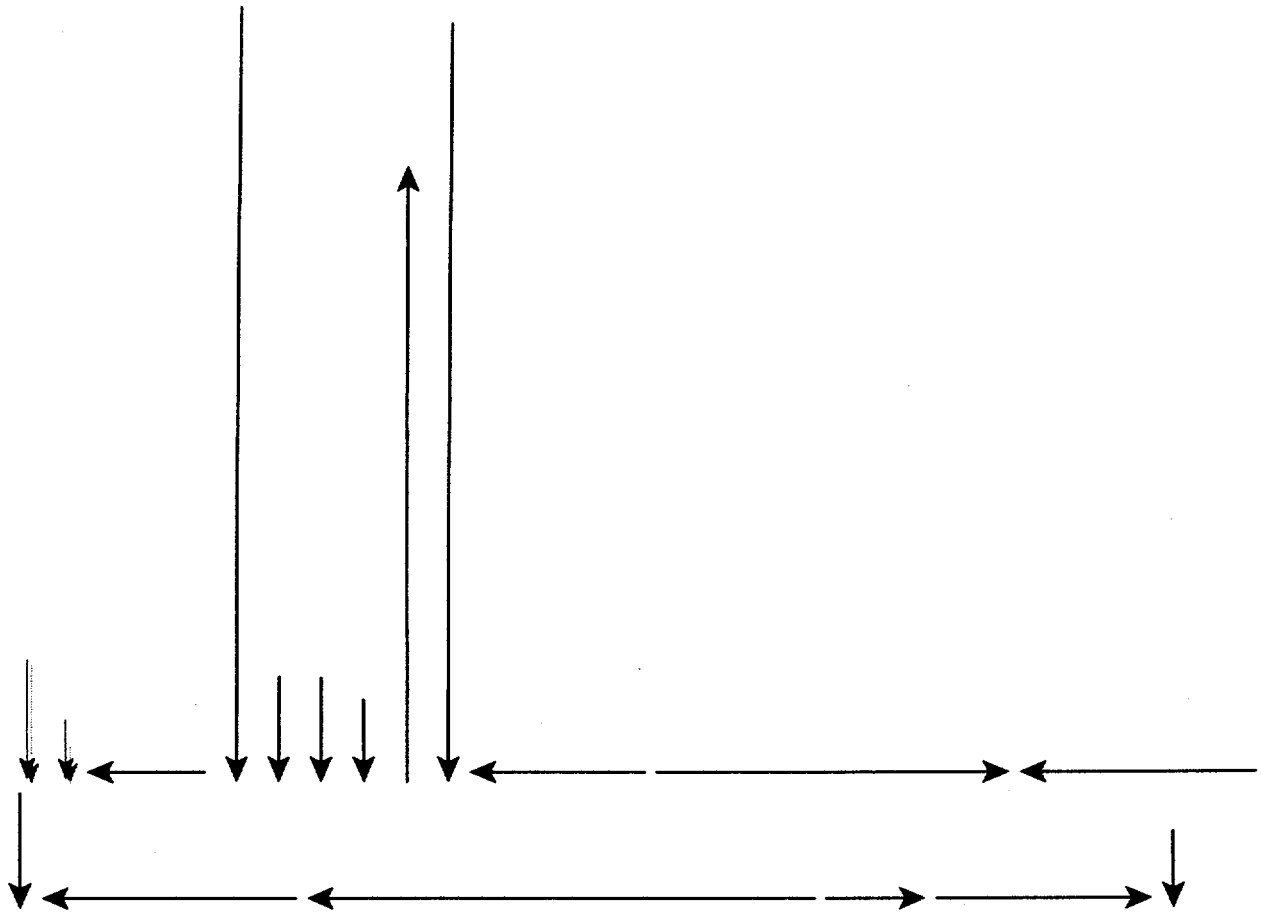
Given the above process, a high degree of confidence should exist that a serious smoke infiltration event can be mitigated without increased risk to the public.

Attachment 1 seems to end with staying or leaving the control room but does not present much guidance on when both the control room and the remote shutdown areas are challenged.

ATTACHMENT 1

Attachment should be sufficiently broad in scope to address both a fire which is external to the control room and one which challenges the remote shutdown capability.

Note: Numbers in ( ) correspond to description in text



Should there be a decision block here which asks the question whether the remote shutdown locations are available and whether the preferred pathway to the remote shutdown locations are available?

Should this item appear after the attempts to purge smoke successful appears?

Success does not occur unless the reactor can be shutdown from the remote shutdown panel.

Proposed Agenda for June 14 and June 15, 2000
Control Room Habitability Analysis Subgroup

Purpose

- Continue Discussion on Control Room Habitability analyses at nuclear power plants

PROPOSED AGENDA

June 14, 2000 8:00 AM – 12:00 PM

Item	Topic	Responsible Party
1	Welcome and Opening remarks	NRC Liaison /TF Subgroup chair
2	Summary of accomplishments at the May meetings, with notes of interim progress	NRC Liaison /TF Subgroup chair
3	Review of Revised Active Action Items Listing	All
4	Discussion of Issues/Action Items (See Active Actions):	All
	♦ Valve Discharge Performance Info for Atmospheric Dispersion Modeling Improvement (AI 92)	
	♦ Elevated Releases with ARCON 96 (AI 25)	
	♦ Containment Mixing Rates Write-up (AI 7)	
	♦ Update of PWR MSLB Write-up (AI 114) -- Format and Content Update	
	♦ PWR Locked Rotor Analysis Write-up (AI 95)	
	♦ BWR SLB Analysis Status (AI 116)	
5	Adjust second day agenda, as appropriate	NRC Liaison /TF Subgroup chair

Analysis Subgroup

PROPOSED AGENDA

June 15, 2000 8:00 AM – 12:00 PM

Item	Topic	Responsible Party
1	Introductions and Review First Day's results.	NRC Liaison /TF Subgroup chair
2	Continue Discussion of Issues/Action Items (See Active Actions): ♦ BWR MHA LOCA Analysis Write-up (AI 149) ♦ PWR MHA LOCA Analysis Write-up (AI 148) ♦ BWR Control Rod Drop Write-up (AI 118) ♦ Guidance for Use of SCBAs for Radiological Events (AI 147)	All
3	Review of Revised Active Action Items Listing	All
4	New Business	All
5	Define topics for next NEI/NRC subgroup meeting. Define future actions and schedule.	NRC Liaison /TF Subgroup chair

NEI Control Room Habitability Task Force Position on Containment Mixing Rates

A containment mixing rate (i.e., a mixing rate between the sprayed and unsprayed regions of the containment) can be determined based on the cooldown rate in the sprayed region and the buoyancy-driven flow that results. Such a mixing rate is conservative in that it does not take credit for other mixing mechanisms which would affect the unsprayed region including momentum exchange between the spray droplets and the sprayed region atmosphere, flow from the unsprayed region to the sprayed region due to steam condensation in the sprayed region, and the effect of heat sources in the unsprayed region (i.e., heat sinks that become heat sources during cooldown).

Consider a containment which is initially well-mixed (consistent with the assumption of a uniform distribution of fission products). Consider a cooldown rate in the sprayed region, dT/dt, a mixing rate, λ (expressed as a multiple of the unsprayed region volume per unit time), and a temperature difference between the sprayed and the unsprayed region, δT. The steady state relationship (based on an energy balance) is:

$$\lambda = dT/dt/\delta T \qquad \text{Eqn 1}$$

where dT/dt will also be the cooldown rate in the unsprayed region.

The other relationship that defines the two unknowns in Equation 1 (λ and δT) is the one describing the buoyancy-driven flow through some limiting effective flow area $A/\sqrt{K}$. With a driving head, H, and a unsprayed volume, V_u, the relationship is as follows:

$$\lambda = A/\sqrt{K}(\sqrt{2gH\delta T/T})/V_u \qquad \text{Eqn 2}$$

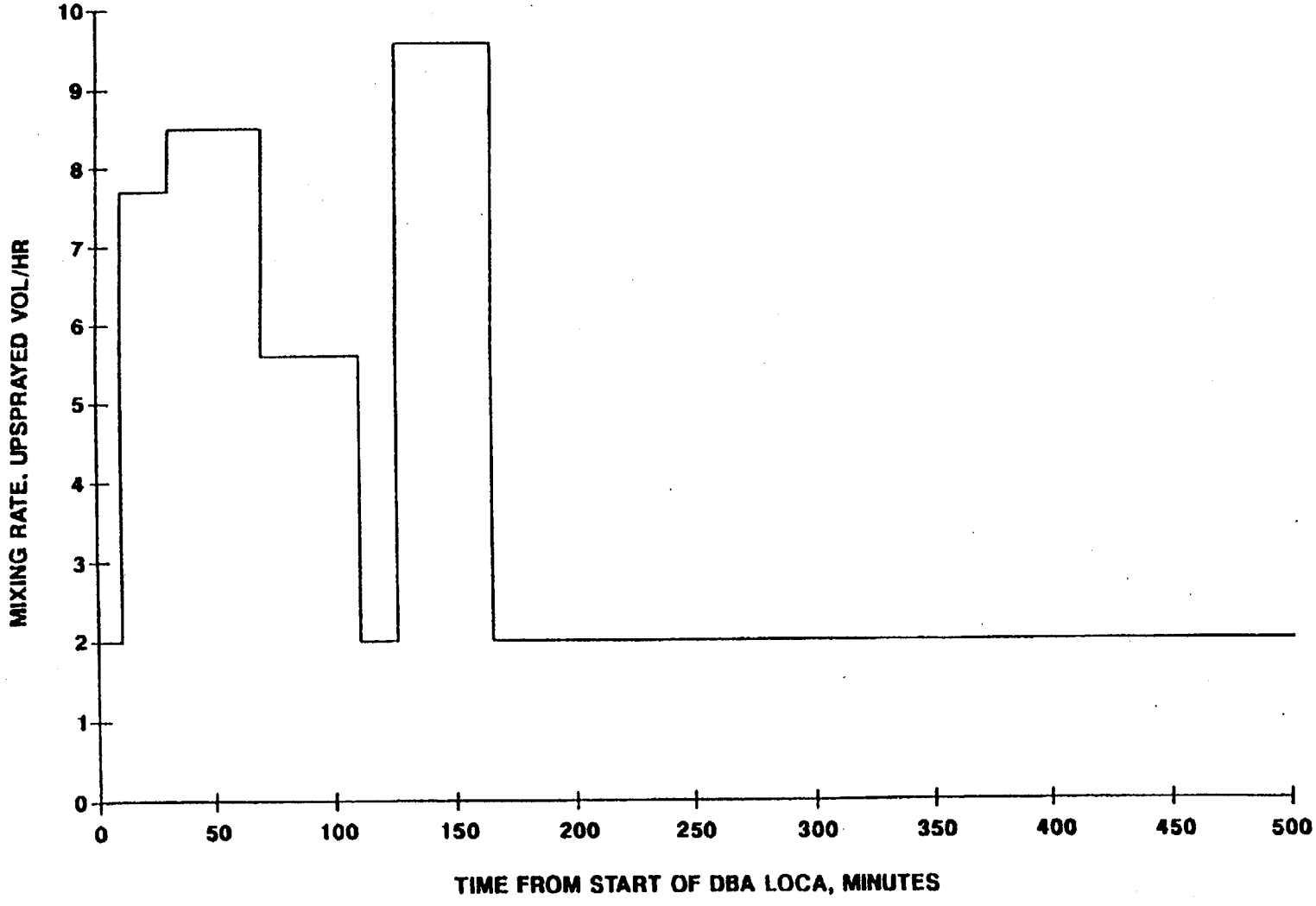
where g is the acceleration due to gravity.

For a typical PWR, $A/\sqrt{K} = 1000$ ft^2, H = 25 ft, and $V_u = 5.5E5$ ft^3. For these typical geometric values and a sprayed region cooldown rate of 30 F/hr, Equations 1 and 2 yield a mixing rate of approximately 15 unsprayed volumes per hour. For a cooldown rate of 10 F/hr, the mixing rate would be approximately 10 unsprayed volumes per hour.

The model described above (and the typical PWR plant parameters cited) appear in the "Source Term Update for the Evolutionary Advanced Light Water Reactor" prepared by the Advanced Reactor Severe Accident Program (ARSAP) Source Term Expert Group (September 1990). The model was applied to the Design Certification of System 80+, and that application is described in Section 6.5 of CESSAR-DC. Figure 6.5-4 of CESSAR-DC shows the containment mixing rate for System 80+ as a function of time calculated using the above model.

NRC is encouraged to endorse the above modeling concept. It would be incumbent on any licensee applying this model to identify the geometric input ($A/\sqrt{K}$, H, and V_u) and the appropriate cooldown rate(s). The current Standard Review Plan mixing rate of two

unsprayed volumes per hour would be considered a minimum value and would be permitted for use (1) by those licensees not choosing to apply the above model or (2) at times when the cooldown rate giving a value greater than two per hour is not defined or does not exist for a given plant.



DBA LOCA Containment Mixing Rate

Figure 6.5-4

Pg 3

OUTLINE OF MAIN STEAM LINE BREAK ACCIDENT ANALYSIS FOR PWRS:
Jerry Gryczkowski - BGE

Definition: Pre-trip guillotine-type rupture of a main steam line outside containment between steam generator and Main Steam Isolation Valve (MSIV)

Sequence of Events:
- (a) Pre-trip guillotine-type rupture of a main steam line outside containment between steam generator and MSIV
  - (i) A spectrum of MSLB sizes should be analyzed.
  - (ii) Initiation of a MSLB from HFP and HZP should be analyzed.
- (b) Increased rate of heat extraction by the affected steam generator
- (c) Cooldown and depressurization of the reactor coolant system (RCS)
- (d) RCS cooldown causes a positive reactivity addition with a negative MTC and FTC, causing core power level and heat flux to increase.
- (e) Steam generator depressurization initiates a reactor/turbine trip on low steam generator pressure
- (f) Steam generator depressurization causes MSIVs to trip closed
- (g) Positive reactivity addition terminated on CEA insertion post-SIAS
- (h) HPSI pumps started on SIAS, however, RCS pressure too high for safety injection flow
- (i) RCS pressure continues to decrease and Pressurizer empties
- (j) RCS pressure drops to HPSI shutoff head and safety injection flow begins
- (k) HPSI adds sufficient coolant mass to reestablish Pressurizer level
- (l) HPSI adds more negative reactivity
- (m) Turbine trip causes loss of AC (LOAC)
- (n) LOAC causes reactor coolant pumps (RCPs) to coast down, minimizing core flow, lowering DNBR, maximizing failed fuel pins. The analysis should be performed with and without a LOAC.
- (o) Blowdown of the affected steam generator continues till dry.
- (p) Cooldown of the RCS is terminated when the affected SG blows dry and AFW flow is isolated to the ruptured SG.

Source Terms with Fuel Failure:
- (a) Per SRP 15.1.5 App.A, "if the MSLB accident is predicted to cause such fuel failure, a dose analysis will be performed with the corresponding iodine activity but without a concurrent iodine spike."
- (b) Initial thermal power is the UFSAR rated thermal power times a factor of 1.02 per Regulatory Guide 1.49.
- (c) The power peaking factor is per the COLR/Technical Specifications.
- (d) The isotopic source terms per unit thermal power are per TID-14844.
- (e) 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.
- (f) To maximize fuel failure, the maximum-worth control rod should be assumed to be held in the fully withdrawn position per SRP 15.1.5.
- (g) The gas gap fractions are per Reg. Guide 1.25 and any associated SERs.
- (h) The I-131 DEQ specific activity of the Secondary Coolant System is that specified in the Tech Specs. (Typically 0.1 μCi/gm)
- (i) The I-131 DEQ specific activity of the Primary Coolant System is that specified in the Tech Specs. (Typically 1.0 μCi/gm)
- (j) The noble gas specific activity of the Primary Coolant is that specified in the Tech Specs. (Typically 100/<E> μCi/gm, where <E> is the sum of the average gamma and beta energies emitted by the xenon and krypton isotopes)
- (k) The minimum RCS fluid mass should be assumed.
- (l) The maximum secondary fluid mass should be assumed.
- (m) The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the RCS.

Source Terms with No Fuel Failure and a Preaccident Iodine Spike
- (a) Per SRP 15.1.5 App.A, if the MSLB accident is predicted to not cause fuel failure, a dose analysis will be performed with the corresponding iodine activity and with a preaccident iodine spike."
- (b) The I-131 DEQ specific activity of the Secondary Coolant System is that specified in the Tech Specs. (Typically 0.1 μCi/gm)
- (c) A reactor transient has occurred prior to the postulated MSLB and has raised the RCS I-131 DEQ concentration to the maximum value permitted by the Tech Specs (typically 60 μCi/gm)

(d) The noble gas specific activity of the Primary Coolant is that specified in the Tech Specs (typically 100/<E> μCi/gm, where <E> is the sum of the average gamma and beta energies emitted by the xenon and krypton isotopes)
(e) The RCS fluid mass, which maximizes dose, should be assumed.
(f) The maximum secondary fluid mass should be assumed.

Source Terms with No Fuel Failure and a Concurrent Iodine Spike
(a) Per SRP 15.1.5 App.A, if the MSLB accident is predicted to not cause fuel failure, a dose analysis will be performed with the corresponding iodine activity and with a concurrent iodine spike."
(b) The I-131 DEQ specific activity of the Secondary Coolant System is that specified in the Tech Specs. (Typically 0.1 μCi/gm)
(c) 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 Tech Specs (Typically 1.0 μCi/gm)
(d) The noble gas specific activity of the Primary Coolant is that specified in the Tech Specs (typically 100/<E> μCi/gm, where <E> is the sum of the average gamma and beta energies emitted by the xenon and krypton isotopes)
(e) The RCS fluid mass that maximizes dose should be assumed.
(f) The maximum secondary fluid mass should be assumed.
(g) The concurrent iodine spike duration is 8 hours per DG-1081 or per the licensing basis.

Primary-to-Secondary Leak Rate:
(a) The primary to secondary leak rate is per Tech. Spec. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.
(b) The density used in converting volumetric leak rates to mass leak rates should be consistent with the procedures used to calculate the leak rate. In most cases, cooled liquid is assumed (1.0 gm/cc or 62.4 lbm/ft^3).
(c) Primary-to-secondary leakage to the affected steam generator should be assumed to continue until the RCS temperature is less than 100°C or per the licensing basis.
(d) Primary-to-secondary leakage to the unaffected steam generator should be assumed to continue until the primary system pressure is less than the secondary system pressure. The release of radioactivity from the unaffected steam generator should be assumed to continue until shutdown cooling is in operation or per the licensing basis..

Secondary-to-Atmosphere Transport:
(a) All noble gases released from the RCS are assumed to be released to the environment without reduction or mitigation.
(b) All iodines released from the RCS via the affected steam generator should be assumed to be released to the environment without reduction or mitigation.
(c) An expansion model of the affected steam generator blowdown may be assumed.
(d) All iodines released from the RCS via the unaffected steam generator with tubes uncovered should be assumed to be released to the environment without reduction or mitigation.
(e) Iodines released from the RCS via the unaffected steam generator with tubes submerged should be assumed to mix with the bulk water in the steam generators. This leakage is released to the environment at a rate based on the steam mass flow rate from the steam generator. A partition coefficient of 100 should be assumed.
(f) The Westinghouse Owners Group has its own proprietary model.

Alternate Repair Criteria:
Determine the primary-to secondary leakrate and the primary I-131 DEQ concentration from the flex methodology of DG-1074.

Atmospheric Transport to the Site Boundaries and to the Control Room:
(a) The 2 hour Exclusion Area Boundary (EAB) Atmospheric Dispersion Coefficient (X/Q) values should be calculated per SRP 2.3.4. The results are usually detailed in the UFSAR.
(b) The 30 day Low Population Zone (LPZ) X/Q values should be calculated per SRP 2.3.4. The results are usually detailed in the UFSAR.
(c) The 30 day control room X/Q values should be calculated by either Murphy-Campe or NUREG-CR/6331 Rev.1 methodology. A distinct set of X/Q values must be calculated and utilized for each source-receptor pair e.g. Main Steam Gooseneck to the control room, ADV to the control room,...)
(d) Per SRP 15.1.5, a Loss-of-Offsite-Power (LOOP) should be assumed, which does not allow use of the condensers.

(e) No credit should be taken for deposition of the plume on the ground or decay of isotopes in transit to the site boundary or control room.

Control Room Inputs:

(a) The control room volume should be extracted from the design basis.

(b) The control room time-dependent unfiltered inleakage should be determined.

- (i) Prior to control room isolation, maximum inflow should be used.
- (ii) A conservative time to isolation should be determined via testing and as a function of manual/automatic closure.
- (iii) After control room isolation, inleakage should be determined via appropriate testing and per AOP.

(c) The control room filtered inflow should be extracted from the design basis. The relevant filter efficiencies should be per Tech. Spec.

(d) The breathing rate for control room personnel should be per Reg. Guide 1.4 (3.47E-04 m^3/sec)

(e) The control room occupancy factors are extracted from SRP 6.4:

(f) The control room recirculation and filtration may be credited.

- (i) Flow should be the Tech. Spec. minimum.
- (ii) The filtration efficiencies should be those listed in the Tech. Specs. A safety factor of two must be included per GL-99-02.
- (iii) Filtration initiation delay time based on SIAS activation time, emergency diesel generator startup time, and a margin term should be modeled.

EAB and LPZ Inputs:

The breathing rates for offsite dose calculations should be per reg. Guide 1.4:

Miscellaneous Inputs:

(a) Per SRP 15.1.5, the worst single active component failure should be assumed to occur (e.g. the failure of a EDG and thus a complete train of safety-related equipment)

(b) Per SRP 15.1.5, only safety grade equipment should be assumed operative to mitigate the consequences of an accident.

(c) ICRP 30 dose conversion factors should be utilized.

(d) Buildup of daughter nuclides need not taken into account as source term nuclides decay per Reg. Guide 1.25.

Acceptance Criteria and JCOs:

(a) The SB and LPZ offsite dose limits are defined per SRP 15.1.5.

(b) The control room dose limits are defined per 10 CFR 50 Appendix A GDC 19 and per SRP 6.4.

(c) Upon a failure to meet regulatory requirements, a JCO relying on KI tablets and SCBAs may be utilized. The criterion to utilize KI tablets and SCBAs in JCO determinations may be incorporated in the FSAR.

An Outline of The Dose Consequences Analysis for The Reactor Coolant Pump Rotor Seizure and Reactor Coolant Shaft Break (PWR) for the NEI CRH Task Force Report NEI-99-03 Appendix - E

Syed A. Ahmed, Dominion Generation.

E.1 Sequence Of Events

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.

Some fuel damage is 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.

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 (WCAP-13132).

E.2 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-131and 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.

E.3 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.

E.4 Control Room Design and Ventilation

Determine the volume of the control room envelope. The control room ventilation design should be considered i.e., normal and emergency ventilation intake location (to be used for the Chi/q calculations together with release point and met data) and ventilation flow rate (CFM) to the control room, automatic or manual control room isolation (review of EOP).

Note that depending on system design a control room isolation signal may not be generated after the accident and may require manual isolation of the control room. In this case a delayed isolation of the control room would have to be considered. Also, some control rooms are equipped with a recirculation and filtration system if such feature exits it should be considered. Determine the control room emergency ESF ventilation filter design (2 in. vs. 4 in.) and whether they are equipped with heaters. Establish filter efficiency based on Technical Specifications or other criteria such as testing.

Establish an inleakage assumption. Consideration of whether the control room is pressurized vs. un-pressurized and other design features of the control room should be included.

E.5 Other Input Parameters

1) Chi/q Values based on PORV release point and several years of met data and inlet to control room (Reg. Guide1.145, ARCON96),
2) Dose conversion Factor from ICRP30,
3) Occupancy Factor ,
4) Breathing Rate,
5) Ventilation Flow Rate,
6) Steam and Liquid Flow Rate through PORV,
7) Duration: Effected generator with stuck PORV, release stops within 30 minutes, Cycling release stopped in 8 hours or less. Note that the control room doses are calculated for 30 days.

E.6 References

1) NUREG-800 SRP Section 15.3.3-15.3,
2) NUREG-800 SRP Section 6.4,
3) NUREG/CR-6331, Rev. 1, "Atmospheric Relative Concentrations in Building Wakes, ARCON96, USNC 1997.
4) 10 CFR 50 Appendix A GDC 19,
5) 10 CFR 100,
6) Reg. Guide 1.25,
7) Reg. Guide1.145,
8) Reg. Guide 1.52,
9) Reg. Guide 1.4,
10) International Commission Radiological Protection (ICRP) Publication 30, Pergammon Press, 1982.

CRH Task Force
Analysis Sub-Group
Topic S-09

Action # 147: Develop basis for taking credit for use of SCBAs for radiological events.

NRC Position: Use of SCBAs and KI is acceptable for addressing CRE integrity in the interim situation until the issue is remediated. However, use of SCBAs or KI in the mitigation of situations where in-leakage does not meet design basis limits is not acceptable as a permanent solution.

CRH Position: Long term credit should be provided on a case by case basis (if currently part of the licensing basis). Otherwise, a basis for application needs to be developed.

It is the opinion of the CRH Task Force that 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, several points must be addressed by licensees. 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.

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 durations 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, surveilled, 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 durations 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 programs that meet 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.

QUESTION FOR TASK FORCE:
Would it be possible for those licensees who have converted their licensing basis to the Alternative Source Term (AST) to receive an exemption (from 10 CFR 20, RG 8.15, NUREG-0041?) which will allow the use of particulate filter respirators for radiological emergencies based on the change in iodine chemical form (e.g., 91% aerosol iodine)? Filter type respirators are less stressful (both physically and psychologically) in their use and would allow more flexibility for use over a longer time period without having to change air supply bottles.

Background and Supporting Information:

The following documents were reviewed:

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

CONTROL ROD DROP ACCIDENT ANALYSIS FOR BWRs:

Accident Description:

The control rod drop accident 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:

1. The combination of reactor operating mode, control rod positions, core burnup, etc., that results in the largest source term is selected for evaluation.
2. No allowance is made for activity decay prior to accident initiation, regardless of the reactor status for the selected case.
3. The amount of radioactivity accumulated in the fuel-clad gap is assumed to be the same as that in Regulatory Guide 1.77.
4. 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 which propose that fuel melting is not likely to result in significant releases prior to MSIV closure.
5. 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.
6. 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.
7. 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:

1. A coincident loss of offsite power is assumed at the time of the accident.
2. The integrity of the turbine and condenser is unaffected by the rod drop accident.
3. The atmospheric dispersion factors (X/Q), breathing rates, and dose conversion factors are the same as those used in the calculation of doses from a LOCA.

Activity transport to the atmosphere:

1. All noble gases remain in a gaseous state and are available for leakage from the turbine and condensers.
2. Of the iodines which reach the turbine and condensers, 90% are removed by partitioning and plateout in the turbine and condensers leaving 10% airborne and available for leakage.
3. 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.
4. The effects of radiological decay during holdup in the turbine and condensers are taken into account.

Control Room Inputs:

1. The control room volume should be that as described in the plant's design basis.
2. The control room time-dependent unfiltered inleakage should be determined.
3. Prior to control room isolation, maximum inflow should be used. A conservative time to isolation should be determined via testing. After control room isolation, inleakage should be determined.
4. The control room filtered inflow should be taken from the design basis. The relevant filter efficiencies should be per Tech. Spec.
5. Standard breathing rates and occupancy factors per Reg. Guide 1.3/1.4 and SRP 6.4 respectively should be used. These will be listed in the plant's design basis.
6. Control room recirculation and filtration may be credited. Flow should be at the Tech Spec minimum.

Accident Duration:

24 hours unless demonstrated shorter by plant design.

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.

MAXIMUM HYPOTHETICAL ACCIDENT ANALYSIS FOR BWRs:

Accident Description:

The Maximum Hypothetical Accident (MHA) is the accident that produces the maximum amount of fuel damage. For most plants, this is the Loss of Coolant Accident (LOCA). 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 large-break 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 LOCA are treated separately and then summed to obtain a total dose. These dose contributions are due to containment leakage (including the contribution from containment purge valves during closure), post-LOCA leakage from ESF systems outside containment, MSIV Leakage, and shine from sources outside the control room envelope.

The radiological consequence analysis described here is based on the use of TID-14844 and Regulatory Guide 1.3. The use of Alternative Source Terms (AST) is described in NUREG-1465 and Draft Regulatory Guide 1081. AST is not specifically addressed here.

Core Source Term Assumptions:

1. The initial power level used in the analysis is taken as the licensed core thermal power plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified.
2. 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 suppression pool free air volume may be included provided there is a mechanism to assure mixing between the drywell to the wetwell.
3. The iodine released to the containment is assumed to be composed of 91% elemental, 4% organic, and 5% particulate.

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 which 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.

1. The primary containment leakage rate is assumed to remain constant over the course of the accident. However, based on discussions with the NRC, this leakage may remain constant at one half the initial leak rate after 24 hours for a BWR (similar as that for a PWR).
2. 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.

3. Noble gas releases to the environment are unaffected by the presence of filters or sprays.
4. Reduction in airborne radioactivity in the containment 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.
5. Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02.
6. 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.
7. Where dilution credit for a secondary containment with recirculation is claimed, adequate mixing in the secondary containment volume should be demonstrated.
8. 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.

1. 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.
2. For plants that do not provide an ESF atmosphere filtration system, the dose assessment should include the leakage from a gross failure of a passive component. This leakage should conservatively be assumed to be 50 gallons per minute, starting at 24 hours after the accident and lasting for 30 minutes. However, the NRC has agreed to forego this requirement provided that the ESF leakage used in the radiological consequence analysis is double that of the value specified in technical specifications.
3. 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.
4. 50% of the core iodine inventory should be assumed to be mixed in the sump water being circulated through the containment external piping systems.
5. 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.

6. 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.

1. 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.
2. 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.
3. No release of activity from the MSIV Leakage Control System (MSIVLCS) is assumed up to the time of system actuation.
4. 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.
5. MSIV releases which 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.
6. 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.
7. 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.
8. 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.
9. 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.
10. Holdup and dilution in the turbine building should not be assumed.

Miscellaneous Input parameters, Assumptions, and Initial Conditions:

1. An adequate failure mode analysis has been performed to justify the selection of the most limiting single active failure for use in the radiological consequence analysis. A coincident loss of offsite power is assumed at the time of the accident.
2. Standard breathing rates and occupancy factors are per Reg. Guide 1.3. Standard occupancy factors are per SRP 6.4. Other values will be considered on an individual case basis. These should be consistent with those described in the plant's design basis.
3. Dose conversion factors used in the analyses, should be consistent with those described in the plant's design basis. However, the use of ICRP 30 dose conversion factors are recommended, as these may provide lower calculated doses.
4. The atmospheric dispersion factors (X/Qs) used in the analysis are the same as those listed in the plant's design basis as applicable to each release point. Credit for an elevated release should be assumed only if the release height is more than two and one-half times the height of any adjacent structure.
5. 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.

Control Room Habitability Analysis Inputs:

Control room habitability dose analysis is based on a summation of doses resulting from primary containment leakage, MSIV leakage, ESF leakage, and direct shine from outside the control room envelope.

1. The control envelope volume should be that as described in the plant's design basis.
2. The control room filtered inflow should be taken from the design basis. When using design ventilation flow rates, the measurement uncertainties that would result in the highest calculated dose should be considered.
3. The control room time-dependent unfiltered inleakage, considering appropriate uncertainties, should be determined. Prior to control room isolation, maximum inflow should be used. A conservative time to isolation should be determined via appropriate testing. After control room isolation, a maximum amount of unfiltered inleakage is assumed and should be demonstrated via appropriate testing.
4. The relevant filter efficiencies used in the analysis should be per technical specifications.
5. Standard breathing rates and occupancy factors per Reg. Guide 1.3 and SRP 6.4 respectively should be used. Justification for other values will be considered on an individual case basis. These values will be listed in the plant's design basis.
6. Control room recirculation and filtration may be credited. Recirculation flow should be at the technical specification minimum.
7. The control room operator dose due to direct shine from outside sources must also be included in the total dose evaluation.

Meteorology and Atmospheric Dispersion Assumptions:

1. Atmospheric dispersion values (X/Q) for the EAB, LPZ, and the control room that were approved by the NRC staff during initial facility licensing or in subsequent licensing proceedings may be used in performing radiological analyses.
2. Methodologies that have been used for determining X/Q values are documented in Regulatory Guides 1.3, 1.145, and the paper "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (K.G. Murphy and K.W. Campe, August 1974).

Accident Duration:

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

Acceptance Criteria:

Calculated doses to control room operators must be within the requirements in 10 CFR Part 50, Appendix A, General Design Criterion 19 (GDC 19). This includes 5 rem whole body or its equivalent to any organ. SRP 6.4 defines this organ dose to be 30 rem to the thyroid. However, per discussions with the NRC staff, thyroid doses as high as 50 Rem will be acceptable on an individual case basis.

The offsite 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 LOCA if the calculated whole body and thyroid doses at the EAB and LPZ are less than the exposure guideline values in 10 CFR Part 100.

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NOTE:
THE FOLLOWING SECTION IS SUBJECT TO CHANGE. WORK IS PENDING REGARDING CHANGES TO THE ARCON96 SOFTWARE.

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Use of NRC Computer Code ARCON96:

The ARCON96 code contains processing options that may yield X/Q values that are not sufficiently conservative for use in accident consequence assessments or may be incompatible with release points and ventilation intake configurations at particular sites. The applicability of these options and associated input parameters should be evaluated on a case-by-case basis.

The May 9, 1997 version of the NRC computer code ARCON96 as described in NUREG/CR-6331, R1 is an acceptable methodology for assessing control room X/Q values for use in design basis accident radiological analyses, subject to the conditions listed below, unless unusual siting, building arrangement, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions indicate otherwise.

1. The ARCON96 code must be obtained and maintained under an appropriate software quality assurance program that complies with the applicable criteria of 10 CFR Part 50 Appendix B and meets other applicable industry consensus standards. Although the software was developed under a software quality assurance program, the licensee is ultimately responsible for the accuracy and appropriateness of use of the ARCON96 results.

2. Meteorological observation data input into ARCON96 are obtained from instrumentation that are maintained under the site's meteorological measurements program, as described in the facility's licensing basis. The data must be shown to be representative for the control room X/Q assessments. Three 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.

3. 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 systems response or manual operator actions. While ventilation intakes can be located via reviews of FSAR drawings, the location of significant infiltration intakes is more subjective and will require judgement on the part of the dose analyst.

   a) A X/Q value should be evaluated for each release-intake combination. It may be possible to qualitatively show that the X/Q values for some release-intake combinations would be bounded by values calculated for other combinations, reducing the number of needed calculations.

   b) The licensee should use the most restrictive (i.e., resulting in the highest dose) X/Q value for each release-intake combination applicable to the particular radiological analysis.

   c) For control rooms with dual intake designs, the guidance of Section III.D and Figure 1 of the Murphy-Campe paper applies. Also, the practice of determining the X/Q 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.

4. ARCON96 provides options that allow a user to model three different release types - ground level, stack, and vent. An area source can be modeled as a subtype of a ground level release.

a) Ground Level Release: The ground level release type is appropriate for the majority of control room X/Q assessments.

b) Stack Release: The stack release type is appropriate for releases from standalone stacks that are two and one-half times the height of adjacent solid structures. Plume rise ... ... Use of the elevated plume option may lead to unrealistically low concentrations at control room intakes located close to the base of tall stacks. If the X/Q values generated by ARCON96 are all extremely low, other models should be used to estimate the potential control room intake X/Qs during low wind speed conditions.

   i) If addressed in the current licensing basis, fumigation conditions are to be considered using the guidance of Regulatory positions 1.3.2.b, 2.1.2, and 2.2.2 of Regulatory Guide 1.145. Ground level X/Q values generated by ARCON96 may be substituted for values generated with equation 5 of RG 1.145.

c) Area Source: The diffusion models in ARCON96 are based on point-source formulations. However, some release sources can be better characterized as area sources. Examples might include postulated releases from the surface of a reactor or secondary containment building, or releases from multiple points such as the roof vents on typical turbine buildings. ARCON96 reduces an area source to a virtual point source using two initial diffusion coefficients entered by the code user.

   i) LOCA radiological analyses have typically assumed that the containment structure could leak anywhere on the exposed surface. As such, these analyses typically used the shortest distance between the containment surface and the control room intake and treated the containment as a point source. This approach may be 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.

   ii) Since leakage is more likely to occur at a penetration, dose analysts must consider the potential impact of containment penetrations exposed to the environment within this modeled area. It may be necessary to consider several cases to ensure that the X/Q value for the most limiting location is assigned. Penetrations that are enclosed within safety-related structures need not be considered here.

   iii) In the absence of site-specific empirical data the initial diffusion coefficients are found by:

   $\text{sigma}_y = (\text{Width}_{\text{area source}}) / 6$

   $\text{sigma}_z = (\text{Height}_{\text{area source}}) / 6$

   iv) The height and width of the area source (e.g., the containment surface) are taken as the maximum vertical and horizontal dimensions of the building cross-sectional area perpendicular to the line of sight to the control room intake. The shortest distance from the building surface to the control room intake is used as the source-receptor distance. The release height is set at the point on the surface of the area source that will result in the shortest slant path.

   v) Bypass leakage from secondary containment buildings may be treated in a similar manner.

   vi) Multiple roof vents could be modeled as a horizontal circular area source of a sufficient radius to encompass all of the vents. This treatment would

be acceptable for those configurations in which (1) the vents are arranged in a pattern that approximates a circular area, (2) if no individual vent is significantly closer to the control room intake than the center of the assumed circular area source, and (3) the release rate from each vent is approximately the same. (The degree of significance will depend on the radius of the assumed circle and the proximity of the vent cluster to the control room intake. As the radius decreases or the distance from the cluster to the control room intake increases the less significance the position of any one vent has.) The distance to the receptor is measured from the closest point on the circumference of the assumed circular area source. In the absence of site-specific empirical data the initial diffusion coefficients are found by:

$\text{sigma}_y = (\text{Diameter}_{\text{area source}}) / 6$

$\text{sigma}_z = 0.0$

d) Vent Release: This type release (mixed mode) will no longer be used.

5. Appropriately structured site-specific atmospheric diffusion tests will be considered by the NRC staff as the basis for deviations from this guidance. 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.

6. With regard to review assignments, the dose analysts are expected to characterize the release point, i.e., location, release height, velocity, duct diameter, type of release(e.g., ground, elevated, area), stack flow, release temperature, source dimensions (if diffuse); and characterize the control room intake, i.e., location height, position relative to release point, etc., as applicable. The assigned meteorologist will review the appropriateness of the licensee's data and perform confirming calculations as deemed necessary, using the parameters provided by or confirmed by the dose analyst.

Information Related to Atmospheric Dispersion for PWR Atmospheric Dump Valve (ADV), Power-Operated Relief Valve (PORV), and Main Steam Safety Valve (MSSV) Discharges Associated with Steam Generator Tube Rupture (SGTR) Design Basis Events

1.0 Background

The purpose of this document is to characterize the thermodynamic conditions, mass flow rates, and velocities of steam discharges from ADVs, PORVs, and/or MSSVs for design-basis SGTRs. This characterization can then be used to explore modeling options for the atmospheric dispersion of those discharges.

To cover a range of plant types and vintages, the Analysis Working Group has compiled available information on the following plants:

1. A Two-Loop Westinghouse plant (1973 vintage)
2. A B&W plant (1973 vintage)
3. The ABB-CE ALWR (System 80+) with additional configuration information from a recent ABB-CE plant (1988 vintage)

Some additional information (of a more limited nature) has also been included on an ABB-CE plant of mid-1970s vintage.

For each of these plants a section is provided with the following:

1. A description of the SGTR scenario as analyzed for the plant in question
2. Discharge configuration
3. Discharge data (thermodynamic conditions, mass flow rates, and velocities)

2.0 Westinghouse 2-Loop Plant

2.1. Scenario Description

As analyzed for this plant the SGTR event begins with the complete severance of a single steam generator tube with the reactor at power. If the main condenser remains available, the event would be mitigated via the condenser steam dump which would preclude a sustained release of steam to the environment. However, in the event of a loss of offsite power, loss of condenser vacuum, or loss of the condenser steam dump itself, then steam would be discharged to the environment through the ADVs and the PORVs.

Once the event begins there is an increase in charging pump flow to compensate for the loss of reactor coolant to the secondary side of the affected steam generator. A steam flow/feedwater flow mismatch also exists. Eventually a reactor trip is generated on low pressurizer pressure. Falling pressurizer pressure also initiates safety injection. This signal terminates normal feedwater flow and starts the delivery of auxiliary feedwater. There is one motor-driven auxiliary feedwater pump and one turbine-driven auxiliary feedwater pump assumed to be available for the affected unit.

The combination of auxiliary feedwater flow and the primary-to-secondary flow causes the affected steam generator level to increase more rapidly than that in the unaffected steam generator. Once the affected steam generator has been identified, then it is isolated. Isolation is accomplished in two conceptual steps (cooldown and depressurization), but these steps can be performed concurrently if controlling steam generator level is a problem (i.e., overfilling of the affected steam generator).

If the main condenser is unavailable, then pressure control in the affected steam generator is accomplished with the ADV. The ADV limits the affected steam generator to a pressure of 1005 psig, but is normally operated in Tave mode (i.e., it limits the pressure indirectly by attempting to control the average primary coolant temperature at 547 F). The ADV is able to pass 590,000 lbm/hr at the steam generator full-load pressure of 750 psia.

Cooldown under conditions of no condenser bypass is accomplished with the PORV on the unaffected steam generator. The PORV will pass 405,000 lbm/hour at 1100 psia, or approximately 10 percent of the rated steam flow for the plant. Over the first half-hour of the event, the core decay power will average a few percent of full power, so the PORV on the unaffected steam generator can be used to lower the reactor coolant system temperature and pressure to a value which eliminates any further environmental discharge of steam from the affected steam generator. The operator's objective, therefore, is to isolate the affected steam generator by:

1. Closing the MSIV on the affected steam generator,
2. Terminating auxiliary feedwater flow to the affected steam generator,
3. Verifying the PORV is closed on the affected steam generator, and
4. Decreasing the primary system pressure and temperature to the point where no further transfer of reactor coolant to the affected steam generator can occur.

The removal of decay power from the core after reactor trip is accomplished in part by the safety injection flow and in part by sensible heat transfer to the secondary coolant (i.e., steaming). 120,000 lbm of reactor coolant is calculated to be transferred from the primary to the secondary side of the affected steam generator in the 30 minutes it is assumed to take the operator to isolate that generator. Considering the roughly 70-80 MBTU that is produced in core decay power and power coastdown heat over that period, it is reasonable to assume that a substantial fraction of this residual core power could be transferred to the affected steam generator at a temperature of 547 F with a corresponding change in enthalpy (above that of the injection conditions) of about 480 BTU/lbm. However, the limited steaming that would result from such an energy balance would be non-conservative; hence, the assumption is made in the dose analysis that the entire 120,000 lbm of primary coolant is released from the affected steam generator ADV over that period.

Once the affected steam generator is isolated, the cooldown is accomplished with the PORV on the unaffected steam generator.

2.2 Discharge Configuration

The ADV discharge is located on top of an auxiliary building. The roof of the auxiliary building is 80' above grade. The control room air intake doghouse is located due east of the ADV discharge stack and is about 25 feet from the stack. The ADV discharge stack terminates about 15' above the roof of the building at an elevation approximately 95' above grade. The air intake is about three feet above the building roof, about 83' above grade.

The containment structure dome reaches a height of about 203' above grade. The air intake is located almost due north of the containment structure at a distance of about 125' from the containment centerline. The intake is about 120' lower than the top of the containment, and the top of the ADV discharge stack is about 108' lower than the top of the containment. Since the containment is about 120' in diameter, the air intake is about 60' from the containment shell. Since the ADV discharge stack is about 25' to the west of the air intake, it is approximately 65' from the containment shell.

The discharge consists of a 16" diameter stack with an 80" long chamfer. Therefore, the discharge area of the stack is about five times that of the 16" cross-section (approximately 7.0 ft^2). However, the choke point would still be in the cross-section of the stack (approximately 1.4 ft^2). The momentum flux is more correctly characterized by the cross-section of the stack than by the exit plane of the chamfer because once the flow exits the stack cross-section, ambient air entrainment would already begin.

2.3 Discharge Data

One can assume that the 120,000 lbm is transferred at constant mass flow of 67 lbm/second (240,000 lbm/hour) over the assumed one-half hour period prior to isolation. This is about 40% of the rated flow of the ADV (although the ADV is rated at a lower pressure). Assuming the upstream conditions are saturation at 1005 psig, the enthalpy would be 1191 BTU/lbm, and a constant enthalpy expansion to atmospheric pressure would yield superheated steam at 296 F and at a density of about 0.032 lbm/ft^3. This means the volumetric flow would be approximately 2094 cfs. Using the cross-section of the stack as the basis for the momentum flux calculation, the momentum flux would be approximately 3109 m^4/s^2. The buoyancy flux is 100.9 m^4/s^3.

<u>3.0 B&W Plant</u>

3.1. Scenario Description

The B&W SGTR scenario is one in which a double-ended guillotine break of a steam generator tube occurs at full power. The assumed single failure is that of an emergency feedwater (EFW) control valve failing close which prevents feedwater from reaching the unaffected steam generator. Until this failure is bypassed, decay power is removed only by steaming the affected steam generator. Once the valve is bypassed, steaming of the affected steam generator is still necessary to control level.

Reactor and turbine trip occur manually 20 minutes after that start of the event. The operators detect the EFW failure 10 minutes after the demand for EFW in the unaffected steam generator. An additional 10 minutes is assumed to effect the bypass. Therefore, EFW is not assumed to reach the unaffected steam generator until 23 minutes after reactor trip (43 minutes after the start of the event).

MSSVs on both steam generators begin to discharge after the reactor trip. Both steam generator's MSSVs close (and remain closed) after approximately 6100 seconds.

The cooldown is assumed to begin at 5580 seconds. By 6690 seconds the RCS temperature has reached 532 F (the temperature necessary to meet the required subcooling margins relative to isolating the effected steam generator. Isolation (and an assumed shift change) require 75 minutes. Beyond this point in time, the ADVs on the affected steam generator are opened only to control level.

The cooldown rate for this analysis is 50 F/hr. A plateau is reached at 450 F, about 5900 seconds after the start of the cooldown at 11190 seconds. 125 minutes after this plateau has been reached, the cooldown is resumed to attain the decay heat removal entry temperature of 246 F. The RCS is then further cooled down using a combination of ADVs and steaming until the affected steam generator temperature becomes less than 212 F at 48367 seconds. This is the end of the analysis.

3.2 Discharge Configuration

This plant consists of three units arranged basically in a north-south direction. The Unit 2 MSSVs and ADVs are close to the combined Unit 1 and Unit 2 control room intake to the north and to the Unit 3 control room intake to the south. In all cases the MSSV and ADV release points are about 40 meters from the intakes.

The elevation of the MSSV release point is 11.2 meters; that for the ADV is 13 meters. The elevation of the air intake is 19 meters. The intakes are located about 37 meters northeast of the respective containment centerlines, and the containments are about 39.25 meters high and about 32 meters in diameter.

3.3 Discharge Data

The post-trip MSSV discharges are about 800 lbm/sec and last 30-40 seconds for both steam generators. These discharges involve multiple valves. During the cooldown, both steam generators average about 17-20 lbm/sec (the higher value being that for that affected steam generator) through the ADVs.

The exact details of the discharge configuration for this plant are still being compiled. However, the mass flows are quite comparable to those presented in the next section; and it is suggested that until further information is compiled, the results of the next section be viewed as applicable to this B&W plant in terms of momentum and buoyancy flux for the MSSV and ADV discharges.

4.0 ABB-Combustion Engineering ALWR

4.1. Scenario Description

The SGTR scenario which is analyzed for the ABB-CE Advanced Light Water Reactor (ALWR) is one in which the ADV on the affected steam generator fails to close after the operator opens it to begin the plant cooldown at 2178 seconds after the start of the event. For this SGTR the reactor trip does not occur until 1758 seconds after the start of the event, and the reactor trip occurs as the result of high water level in the affected steam generator. A loss of offsite power is also assumed to occur. The seven minute delay from the reactor trip to the start of the cooldown accounts for the initial operator delay (five minutes) and an additional two minutes for a single, discrete manipulation.

Once then reactor and turbine trips have occurred, the MSSVs open (about four seconds after the trip) to relieve pressure in both the affected and unaffected steam generators. The discharge for both steam generators over the first 80-90 seconds is about 4.2E4 lbm per steam generator, or an average of about 525 lbm per second. Initially only two MSSVs on each steam generator are open, then four MSSVs are open (for less than 20 seconds), and then only two MSSVs are open again. The maximum MSSV flow rate is about 2000 lbm/sec (for eight valves, four per steam generator), corresponding to peak steam generator pressure of 1272 psig. The rated flow of a single MSSV is 264 lbm/sec.

After the initial lift, the MSSVs open twice again, but only on the affected steam generator. The integrated MSSV flow for the affected steam generator prior to successful isolation is 8.6E4 lbm.

After the reactor (and turbine) trip (as well as the last MSSV lift) the operator tries to establish a 100 F/hr cooldown rate by opening one ADV on each of the two steam generators. The ADV on the affected steam generator is assumed to be open about 12 %, the degree of opening necessary to effect a 100 F/hr cooldown at this point in time. Twelve percent of rated flow at 1100 psia is 35 lbm/sec, and at this flow (and assuming an enthalpy change of 1123 BTU/lbm based on an assumed injection enthalpy of 65 BTU/lbm), the power dissipated by this flow would be about 41.5 Mw(t) per steam generator. The decay power is about 66 Mw at this time, so the sensible heat being removed from the system is estimated to be about 20 Mw to effect the 100 F/hr cooldown.

The operator attempts to close the ADV on the affected steam generator at the time the reactor coolant temperature reaches 550 F (about 15 F below saturation temperature corresponding to the lowest MSSV set point). The ADV is then assumed to stick open, and the ADV block valve is not assumed to be closed until 30 minutes after the attempt to close the ADV (i.e., at 5463 seconds or 1.5 hours after the start of the event). Beyond this point in time, the ADV on the affected steam generator is opened only to control level. These openings (using the other, functioning ADV on the affected steam generator) are treated as full, short-duration openings, although the possibility exists that the operator would attempt to establish a steady rate of steaming. The first of these intermittent openings involves a discharge at a steam generator pressure of 420 psia and a

flow rate of 219 lbm/sec, and the last involves a discharge at 150 psia and a flow rate of 79.5 lbm/sec.

Beyond the closing of the ADV block valve at about 1.5 hours into the event, the cooldown rate is assumed to be reduced to about 20 F/hr. This is conservative because it extends the cooldown time to 8 hours before the entry condition to shutdown cooling is reached. Beyond the entry to shutdown cooling, no further steam discharge from either steam generator occurs.

The discharge rate during the 20 F/hr cooldown from the unaffected steam generator averages about 20 lbm/sec (i.e., about 24 Mw). It is not that time-sensitive because neither the decay power (approximately one percent of reactor power or 40 Mw) nor the sensible heat removal rate (about 4-5 Mw) is changing very rapidly at this time. The unaffected steam generator accounts for about one-half of the decay and sensible heat removal over this period of time, while the affected steam generator (using the ADVs to control level) accounts for the other half.

The total steam release from the unaffected steam generator is about 6.28E5 lbm (including both the MSSV and ADV discharge). That released from the affected steam generator is about 7.01E5 lbm for the ADV only, most of that from level control. The MSSV release for the affected steam generator is about 8.6E4 lbm, as noted above.

4.2 Discharge Configuration

Since the ABB-CE ALWR is not a physical plant, no information is available on the detailed configuration of the MSSV and ADV discharge. However, a similar plant is in operation, and from that plant the discharge configurations have been obtained.

The MSSV discharge flow rate is at a maximum when two MSSVs on a single steam line (four MSSVs per steam generator and eight per plant) are discharging at one time. Since the MSSV stacks are located very close to one another, two MSSVs discharging together can be visualized as a single plume.

Each stack is two feet in diameter, giving a combined area of 6.28 ft^2. The MSSV stacks have a 45-degree chamfer, increasing the combined effective discharge area to potentially 8.9 ft^2 or an effective diameter of 3.4' (about one meter) for two valves discharging together). The elevation of each stack exit is 13.7 meters above grade and about 3.7 meters above the roof elevation of the structure in which the MSSVs are housed.

The ADV discharge is through a large air silencer 54" (1.4 meters) in diameter. Its elevation is also 13.7 meters above grade and 3.7 meters above the roof elevation of the structure in which the ADVs are housed.

As noted previously in this section, the preceding data is based on an actual plant. The following data is based on the ABB-CE ALWR layout. The control room air intakes (there are two) are located approximately 35 meters due east from the probable location of the MSSV and ADV discharges, and these discharge locations are about 43 meters due

north and due south of the containment centerline. The elevation of the control room air intakes is about 22.5 meters above grade. The roof of the structure that would house the MSSVs and ADVs is about 19.8 meters above grade, resulting in an assumed stack elevation for the MSSV and ADV discharges about 23.5 meters above grade. The control room air intakes are about 65 meters northeast and southeast of the containment centerline.

The containment is spherical with the top of the sphere at an elevation 53 meters above grade. The sphere has a diameter of 65.85 meters. Approximately one half of the sphere is located below the roof elevation of the surrounding complex of structures. This complex of structures has a roof elevation to the west of the MSSV and ADV discharge locations that is about 30.5 meters above grade and, therefore, about seven meters higher than the assumed MSSV and ADV discharge stack height. The distance from the MSSV and ADV discharge location to the point where the building complex height increases to the 30.5 meter elevation is about 24 meters. Therefore, for a westerly wind, a wake could exist behind this raised portion of the building complex around the containment which could trap effluent from the discharge stacks in the vicinity of the control room air intakes.

4.3 Discharge Data

There are three situations of interest:

MSSV discharge from the affected steam generator immediately after the reactor trip
ADV discharge at moderate pressure from the affected steam generator to control level
ADV discharge at low pressure from the unaffected steam generator at the end of the cooldown.

For the first condition the flow rate would be about 525 lbm/sec through a combined effective discharge area of 8.9 ft^2. Assuming a superheated steam density of 0.032 lbm/ft^3 at atmospheric pressure (see Section 2.3), the volumetric flow rate would be 1.6E4 cfs, and the discharge velocity would have to be supersonic (1800 fps) in order for the steam to be fully expanded at the exit. The sonic velocity of steam at the assumed conditions is about 1500 fps, so full expansion would not have yet taken place, and 1500 fps is the velocity upon which the momentum flux should be based. The momentum flux for the MSSV discharge would then be about 4.63E4 m^4/s^2. The buoyancy flux would be 422.8 m^4/s^3.

For the second condition the flow rate would be about 219 lbm/sec through the 15.9 ft^2 silencer. Assuming the same 0.032 lbm/ft^3 density, the volumetric flow rate would be 6.84E3 cfs, and the discharge velocity would be 430 fps. The momentum flux for the ADV discharge to preclude overfilling of the steam generator at approximately 2.4 hours into the event would then be about 5.68E3 m^4/s^2. The buoyancy flux would be 180.8 m^4/s^3.

For the third and last condition the flow rate would be about 20 lbm/sec through the 15.9 ft^2 silencer. Assuming the same 0.032 lbm/ft^3 density, the volumetric flow rate would be 6.25E2 cfs, and the discharge velocity would be 39.3 fps. The momentum flux for the ADV discharge from the unaffected steam generator just prior to entering shutdown cooling would then be about 47.4 m^4/s^2. The buoyancy flux would be 16.5 m^4/s^3.

4.4 Supplemental Information

Additional information regarding ADV discharge is available for an operating ABB-CE plant, but the details of the SGTR scenario for that plant have not been compiled for inclusion in this document. Nevertheless, the following geometrical data is available:

The Unit 1 ADV stack is 62.4 meters from the control room air intake at 24 degrees from due north (intake to ADV discharge).

The Unit 2 ADV stack is 62.4 meters from the control room air intake at 67 degrees from due north (intake to ADV discharge).

The top of the ADV stack is 17.2 meters above grade; the control room air intake is 10.7 meters above grade.

For the case of the steam generator at 67 psia and steam discharging at 300 F, the exit velocity is 67.5 m/s. This is based on a discharge diameter of 0.835' (0.2545 meters) and a flow rate of 7279 cfm (3.435 m^3/s).

The corresponding momentum flux for this ADV discharge is 57.8 m^4/s^2. The buoyancy flux is 2.32 m^4/s^3.

5.0 Summary

The following is a summary of the momentum and buoyancy flux values presented in this document for steam discharges from MSSVs, ADVs, and PORVs for SGTR events:

Plant	Discharge Type	Momentum Flux	Buoyancy Flux
W 2-Loop	ADV from affected SG prior to isolation	3.11E3 m^4/s^2	1.01E2 m^4/s^3
B&W Plant	Same as ABB-CE	Same as ABB-CE	Same as ABB-CE
ABB-CE ALWR	MSSV from affected SG prior to isolation	4.63E4 m^4/s^2	4.23E2 m^4/s^3
ABB-CE ALWR	ADV from affected SG to prevent overfill	5.68E3 m^4/s^2	1.81E2 m^4/s^3
ABB-CE ALWR	ADV from unaffected SG prior to shutdown cooling	4.74E1 m^4/s^2	1.65E1 m^4/s^3
ABB-CE Plant	ADV - low SG pressure	5.78E1 m^4/s^2	2.32E0 m^4/s^3

PWR MHA	Was	Is
Thyroid limit	30 Rem	50 Rem
X/Q	Murpy-Campe	ARCON96
Passive Failure	Yes (CLB)	No
Iodine Removal in Sec Containment Bypass Pathways	No	Yes
ESF Leakage Partitioning	10%	10%
Containment Mixing	2/hr	?/hr

BWR MHA	Was	Is
Thyroid limit	30 Rem	50 Rem
X/Q	Murpy-Campe	ARCON96
Passive Failure	Yes (CLB)	No
BWR L_A	Full	1/2 After 24 hr
BWR MSIV	Full	1/2 After 24 hr
Iodine Removal in Sec Containment Bypass Pathways	No	Yes
ESF Leakage Partitioning	10%	10%
Containment Mixing	2/hr	?/hr

FHA		
Thyroid limit	30 Rem	50 Rem
X/Q	Murpy-Campe	ARCON96
DF	100	200
GAP Fractions	12 I31	8
	30 Kr 85	10
	10 other halogen	5
	10 other NG	5
	If > 54 GWD/MTU and LHGR > 6.3 recalculate GAP fractions	
Mixing in FB	N	If justified
Peaking factor	1.65	~1.7

SGTR	**was**	**is**
Thyroid limit	30 Rem	50 Rem
X/Q	Murpy-Campe	ARCON96
	Ground or CLB	(elev or ground)
Spiking	500	335
Spiking Duration	Accident Duration	8 hours
		(4 hrs)
GAP Fractions	12 I31	8
	30 Kr 85	10
	10 other halogen	5
	10 other NG	5
Peaking factor	1.65	~1.7

MSLB	**was**	**is**
Thyroid limit	30 Rem	50 Rem
X/Q	Murpy-Campe	ARCON96
	(ground or CLB)	(ground)
Spiking	500	500
Spiking Duration	Accident Duration	8 hours
		(3 hrs)
GAP Fractions	12 I31	8
	30 Kr 85	10
	10 other halogen	5
	10 other NG	5
Peaking factor	1.65	~1.7

LOCKED ROTOR	**was**	**is**
Thyroid limit	30 Rem	50 Rem
X/Q	Murpy-Campe	ARCON96
	Ground or CLB	(elev or ground)
GAP Fractions	12 I31	8
	30 Kr 85	10
	10 other halogen	5
	10 other NG	5
Peaking factor	1.65	~1.7
Fuel failure	DNBR	Enthalpy

CREA	**was**	**is**
Thyroid limit	30 Rem	50 Rem
X/Q	Murpy-Campe	ARCON96
	Ground or CLB	(elev or ground)
GAP Fractions	RG 1.77	
Peaking factor	1.65	~1.7

CRH Analysis Subgroup Active Actions

Topic **S-01** **S-01: Introduction, History, Organization**

Issue **34** **Section 1 Scope** Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
	6/5/2000	152	Review draft of Section 1 (action 46)	Analysis	O	

Topic **S-02** **S-02: Regulatory Requirements & Guidance**

Issue **35** **Section 2 Scope** Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
	6/5/2000	156	Review draft of Section 2 (action 47)	Analysis	O	

Topic **S-03** **S-03: CRH Issues**

Issue **36** **Section 3 Scope** Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
Radvansky,	5/19/2000	124	Review draft of section 3 (action 48)	Analysis	O	

Topic **S-04** **S-04: Licensing Basis**

Issue **37** **Section 4 Scope** Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
	6/5/2000	161	Review draft of Section 4 (action 49)	Analysis	O	

Topic S-05 S-05: CRH Performance Relative to Licensing Basis

Issue 38 Section 5 Scope Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
Black, D	5/19/2000	127	Review draft of section 5 (action 50)	Analysis	O	

Topic S-06 S-06: Consistency of CRH Assessment

Issue 39 **Section 6 Scope** Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
Metcalf, J	5/19/2000	130	Review draft of section 6 (action 51)	Analysis	O	

Topic **S-07** **S-07: Identification of Plant Specific Vulnerabilities**

Issue **40** **Section 7 Scope** Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
Mscisz, T	5/19/2000	133	Review draft of section 7 (action 52)	Analysis	O	
Schultz / Met	6/15/2000	86	Provide input on most limiting accident to DB/Lic subgroup for sec 7	Analysis	O	Work from RG 1.70 and NEI 93-03, Rev 0

Topic **S-08** **S-08: Significance of CRH Findings**

Issue **41** **Section 8 Scope** Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
Ahmed, S	5/19/2000	136	Review draft of section 8 (action 53)	Analysis	O	

Topic S-09 S-09: Solutions to CRH Problems

Issue 61 Mitigating Actions upon Questionable CRE Integrity — Subgroup: Analysis

NRC Position *Use of SCBAs and KI is acceptable for addressing CRE integrity in the interim situation until the issue is remediated. However, use of SCBAs or KI in the mitigation of situations where in-leakage does not meet design basis limits is not acceptable as a permanent solution*

CRH Position *Long term credit should be provided on a case by case basis (if currently part of the licensing basis). Otherwise a basis for application needs to be developed.*

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
Black, D	6/6/2000	122	Develop basis for taking credit for use of KI for radiological events.	Analysis	O	
Mscisz, T	6/6/2000	147	Develop basis for taking credit for use of SCBAsl for radiological events.	Analysis	O	

Issue 42 Section 9 Scope — Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
	6/5/2000	169	Review draft of Section 9 (action 54)	Analysis	O	
Schultz / Met	8/15/2000	83	Provide input to DB/Lic for section 9 of NEI 99-03	Analysis	O	Provide summary descriptions of the guidance developed in Appendix E
Schultz, S		89	For purposes of JCOs, develop input parameter values / conservatisms that are defendable for all accidents.	Analysis	O	

Topic X-B X-B: External Smoke

Issue 24 **Smoke (Internal and External)** Subgroup: Systems

NRC Position *NRC believes that internally and externally generated smoke was not properly addressed in licensee evaluation and NRC reviews. These two challenges have the potential for the most immediate impact upon operations with the most serious ramifications on public health and safety. NRC desires that NEI 99-03 provide guidance to demonstrate that the licensee adequately manages smoke.*

CRH Position *Smoke is sufficiently managed under App R. Some NRC staff have indicated a qualitative approach to the potential impact of smoke intrusion may be acceptable. Industry agrees to address qualitatively.*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
	5/16/2000	123	Document CRH position on the use of conservatisms with respect to the smoke issue.	Analysis	O	Review the White Paper on Smoke and provide comments and analysis insight
	5/16/2000	139	Review qualitative approach to addressing smoke concerns (action 112)	Analysis	O	

Topic **X-D** **X-D: DBA Radiological Design Parameters**

Issue **48** **Appendix D Scope** Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	*Due Date*	*Action Description*	*Subgroup*	*Status*	*Comments*
Cozens, K	60	Prepare Appendix D (DBA Radiological Input Parameters) of 99-03	Analysis	O	Recommend deleting App D and combining with App E CRH Task Force decision required.

Topic X-E X-E: CRH Dose Analysis

Issue 49 Appendix E Scope

Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
Ahmed, S	6/6/2000	95	Prepare App E input for Locked Rotor	Analysis	O	Prepare draft from FHA outline presented to the NRC
Black, D	6/6/2000	110	Prepare App E input for FHA	Analysis	O	Prepare draft from FHA outline presented to the NRC
Cotton, J	6/6/2000	148	Prepare App E input for a PWR MHA	Analysis	O	Prepare draft from FHA outline presented to the NRC
Gryczkowski	6/6/2000	117	Prepare App E input for Rod Ejection	Analysis	O	Prepare draft from FHA outline presented to the NRC
Gryczkowski	6/6/2000	114	Prepare App E input for MSLB	Analysis	O	Prepare draft from FHA outline presented to the NRC
Jha, K	6/6/2000	116	Prepare App E input for BWR SLB	Analysis	O	Prepare draft from FHA outline presented to the NRC
Mscisz, T	6/6/2000	149	Prepare App E input for a BWR MHA	Analysis	O	Prepare draft from FHA outline presented to the NRC
Mscisz, T	6/6/2000	118	Prepare App E input for Control Rod Drop	Analysis	O	Prepare draft from FHA outline presented to the NRC
Schultz, S		61	Prepare Appendix E (CRH Dose Analysis Approach) of 99-03	Analysis	O	Develop outline / define metholodigies for evaluating (action 89)
Schultz, S	6/6/2000	115	Prepare App E input for SGTR	Analysis	O	Prepare draft from FHA outline presented to the NRC
Schultz, S	6/6/2000	119	Prepare App E input for Waste Gas Decay Tank Rupture	Analysis	O	Prepare draft from FHA outline presented to the NRC

Issue 20 Containment mixing credit

Subgroup: Analysis

NRC Position

CRH Position *Additional mixing credit should be achievable for plants with large unsprayed volumes*

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
Metcalf, J	6/6/2000	7	Review reference material and document industry position on	Analysis	O	Examine approach in AST framework document (EPRI, ~1995) and system 80+ CE application.

containment mixing credit

Present information to the NRC

Issue 19 **Control room occupancy factors** Subgroup: Analysis

NRC Position *GDC 19 is not limited to control operators. It would apply to anyone in the control room. In the TMI event, the plant manager was in the control room for four days and people were sleeping in the control room during the recovery period. Staff believes that operators would tend to stay until the plant was in a safe condition. The current occupancy factors are consistent with 10- or 12-hour shifts with a one-hour shift turnover. Staff indicated exigent conditions could also apply, such as personnel being required to stay on site due to severe weather.*

CRH Position *The Design Basis subgroup agreed that the use of operator in the task force scope statement is consistent with that used in GDC 19 and does not include others present in the control room after an accident except those required by Technical Specifications.*

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
		39	Document industry position on control room occupancy factors. Resolve issue.	Analysis	O	The current industry approach should be utilized and supported with practical examples. Propose actual values for time periods; explore any modifications. On hold

Issue 17 **Credit for suppression pool scrubbing** Subgroup: Analysis

NRC Position

CRH Position

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
Patel, G	6/6/2000	6	Document industry position on suppression pool scrubbing	Analysis	O	Address as part of BWR MHA (action 149)

Issue 11 **Enthalpy vs DNBR** Subgroup: Analysis

NRC Position *DG-1081 approach on determining fuel damage via enthalpy vs. DNBR should offer relief for plants for radiological dose analysis. Staff will consider use of enthalpy vs DNBR to determine fuel damage for reactor accidents other than reactivity insertion accidents.*

CRH Position *Concur that this is applicable to TID source term design basis plants.*

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
Cotton, J	5/20/2000	77	Document applicable accidents	Analysis	O	

Schultz, S	5/2/2000	78	Develop understanding of fuel enthalpy acceptance criteria for fuel failure for "non" reactivity insertion accidents.	Analysis	P	Staff seems receptive to the concept, but much additional work will be required to develop an approved method: Reactor systems branch would review this type of submittal; Revision to vendor topical reports will be necessary. Discussed with John Butler on 5/2/00. Additional help from NEI to pursue opportunity with the vendors is needed. White paper developed under action 10.
Schultz, S	6/6/2000	10	Document industry position on use of enthalpy vs DNBR	Analysis	O	Discuss with the NRC on 6/14/00.

Issue 14 Fuel handling accident parameters Subgroup: Analysis

NRC Position *NUREG-5009 assumptions about gap fractions as a function of burnup are being revisited, so consider these assumptions as under revision. NRC staff not in agreement that relationship between radial peaking factor and burnup provides margin, that the available margin on this and other issues has been used up to justify extended burnup*

CRH Position *An approach similar to the final DG-1081 resolution should be justified for TID source term design basis plants.*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
Black	5/2/2000	90	Develop best estimate input parameter values for fuel handling accident	Analysis	P	This information feeds action 110, FHA input to App E scope. Currently on hold - low priority
Schultz, S	5/2/2000	40	Develop industry position on fuel handling accident parameters	Analysis	O	This information feeds action 110, FHA input to App E scope.
Schultz, S	5/2/2000	91	Develop a basis for crediting operator action to isolate the control room following a fual handling accident. Provide details of timing and the sequence of events.	Analysis	O	This information feeds action 110, FHA input to App E scope
Schultz, S	5/2/2000	96	Determine methods for calculating fuel pin pressures for the fuel handling accident to meet RG 1.25 (involve Owners Groups)	Analysis	O	Pool scrubbing requires <1200 psi. Vendors may need to modify methods. They are currently using values >1200 psi as required by RG 1.25. Write a White Paper to document industry position.

Issue 33 Iodine partition factors Subgroup: Analysis

NRC Position *Staff believes that current guidance is sufficient for TID source term application. Review of industry position will require additional NRC resources.*

CRH Position

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
Patel, G	6/6/2000	79	Formulate a draft position for iodine partition factors in application to TID source term (ECCS leakage)	Analysis	O	Draft study results for discussion with staff.

Issue 15 Iodine Spiking

Subgroup: Analysis

NRC Position *Current state of literature supports reducing spiking on SGTR, but no data supports relaxing this for MSLB. Could find no simple relationship between initial iodine levels and spiking factor. Some acknowledgement that spike duration doesn't last forever*

CRH Position *The data available should support a reduction in the iodine spiking factor.*

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
NRC	6/13/2000	151	Provide comments on iodine spiking duration (action 94)	Analysis	O	
Schultz, S	5/2/2000	94	Decide whether to develop a basis for lowering the spiking value (500) for MSLB or reducing the spiking duration.	Analysis	O	Duration values working paper (J Metcalf) complete 5/2/00. Awaiting NRC comments Spiking values will follow DG-1081 guidance. NRC staff making ACRS presentation on 5/18 - more information should be available after that time.
Schultz, S	6/6/2000	93	Review draft NRC guidance regarding flexible TS and DPO resolution to address the iodine spiking issue.	Analysis	O	Information will be added to App E as an option. Develop industry guidance based on NRC guidance.

Issue 10 Passive failure

Subgroup: Analysis

NRC Position *Will consider deletion of requirement to postulate a 50 gpm leak due to passive failure at 24 hours after an MHA (50 gpm seal leak for 30 minutes). There may be trade-offs with assumptions related to ESF system leakage for plants without safety-related filtration systems. (2 x ESF flow value)*

CRH Position *If not in current licensing basis, should not have to amplify current assumtions.*

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
Schultz, S	5/2/2000	42	Develop industry position on accounting for passive failures	Analysis	O	The principal of eliminating the passive failure is accepted. Compensatory assumptions that may be required with the removal of this conservatism need to be examined carefully.

Issue 9 Thyroid dose limit -50 Rem for TID

Subgroup: Analysis

X-E: CRH Dose Analysis

NRC Position *Thyroid limit can be raised to 50 Rem for TID. TEDE may be used in conjunction with the new source term. To use TEDE with TID source term would require rule making or an exemption request.*

CRH Position

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
Cotton, J	5/2/2000	43	Document industry position on the use of 50 Rem for a thyroid dose limit	Analysis	O	Draft complete 4/00 Provided to staff on 5/2/00. NRC 's final position will depend on the resolution of other issues within NEI 99-03. Need to determine how this acceptance criterion becomes part of the package

Issue **26** **Use of Monte Carlo Methods for Struc Uncert Anal** Subgroup: Analysis

NRC Position *Monte Carlo methodologies may be able to be used to develop an acceptable level of conservatism.*

CRH Position *Use of Monte Carlo techniques must be evaluated before suggesting application*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
NRC - Shap	4/26/2000	120	Provide NRC research on SGTR approach to assessing best estimate plus uncertainty methods (Monte Carlo methodology)	Analysis	O	
Schultz, Met		121	Review NRC research on use of Monte Carlo methodologies	Analysis	O	M. Reinhart interest in Monte Carlo evaluation is high, but will not pursue actively until a goal is defined.

Topic X-F X-F: Sample Calculations

Issue 50 **Appendix F Scope** Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
Schultz, S	5/2/2000	62	Prepare Appendix F (Sample Calculations) of 99-03	Analysis	O	Level of detail (how many events to run sample calcs on) being discussed

Topic X-G X-G: Atmospheric Dispersion

Issue 51 Appendix G Scope — Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
Harvey, B	6/6/2000	63	Prepare Appendix G (Atmospheric Dispersion) of 99-03	Analysis	O	Prepare outline

Issue 7 Elevated releases with ARCON 96 — Subgroup: Analysis

NRC Position

CRH Position *ADVs and MSSVs can be treated as elevated releases*

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
Metcalf, J	5/2/2000	92	Develop a basis for treating ADV/MSSV opening as an elevated release.	Analysis	O	Assemble information on operating performance of ADVs/MSSVs
Metcalf, J	5/2/2000	25	Document industry position on evaluating elevated releases with ARCON 96	Analysis	O	Discussed with NRC staff and V. Ramsdell on 4/6, 4/7
NRC	4/26/2000	3	Provide staff position on elevated relaeses with ARCON 96.	Analysis	O	

Issue 8 NRC guidance document on ARCON 96 — Subgroup: Analysis

NRC Position *Use of ARCON 96 is acceptable within the constraints defined in the recent NRC guidance document.*

CRH Position

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
		12	Review NRC guidance document on ARCON 96 and develop industry position on its use	Analysis	O	Comments were provided to NRC and V. Ramsdell on 4/6/00. See Doc Basis field.

Issue 6 Wind tunnel data — Subgroup: Analysis

X-G: Atmospheric Dispersion

NRC Position *More information is required before the staff will accept the use of wind tunnel data to justify lower X/Qs.*

CRH Position *Wind tunnel experiments should be able to be used to justify lower X/Qs*

Actions:

Responsib	Due Date	Action	Description	Subgroup	Status	Comments
Gryczkowski		26	Document position on new wind tunnel data that can be used in X/Q determination	Analysis	P	Positions discussed with staff on 3/30/00. Based on NRC feedback (more information is needed before this approach can be approved) the issue is on hold.

Topic X-H X-H: Toxic Gas Assessments

Issue 2 **Periodic Reassessment** Subgroup: Systems

NRC Position *External toxic gas challenges need to be periodically reassessed*

CRH Position *The identification and evaluation of nearby potential hazardous gas sources should be updated every three years.*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
Duffy, J	3/22/2000	36	Develop industry position on the need to periodically reassess CR habitability	Analysis	O	Draft completed 3/29/00. See Doc Basis field. Discussed with the NRC on 5/3 - timing should be variable on a plant by plant basis - need final NRC comments Responsibility changed from Systems to DB/L on 5/30/00

Topic **X-J** **X-J: Control Room Envelope Maintenance Program**

Issue **54** **Appendix J Scope** Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
	6/5/2000	176	Provide comments on App J	Analysis	O	

Topic X-K X-K: Technical Specifications

Issue 55 **Appendix K Scope** Subgroup: DB / Lic

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
	5/2/2000	141	Review CRH position on the need for Tech Specs (action 31)	Analysis	O	

Topic **X-O** **X-O: Sample Toxic Gas Calculations**

Issue **59** **Appendix O Scope** Subgroup: Analysis

NRC Position *N/A*

CRH Position *N/A*

Actions:

Responsib	*Due Date*	*Action*	*Description*	*Subgroup*	*Status*	*Comments*
Harvey, B	5/20/2000	100	Prepare Appendix O (Sample Toxic Gas Calculations)	Analysis	O	Development is on hold pending input from Systems and DB/Lic subgroups.
Harvey, B	5/20/2000	80	Determine if the guidance in RG 1.174 and 1.95 are sufficient and if Appendix O is necessary	Analysis	O	

Selected Slides From NRC Presentation to ACRS

Gap--Staff Approach

- ☆ Use NUREG-1465 data for LOCA (0-62 GWD/MTU)
- ☆ Use RG 1.77 data for reactivity insertion accidents
- ☆ Work by PNNL to address environmental impact of change in fuel burnup from 60 to 62 GWD/MTU reported in this period
  - ☆ Will be documented as update to NUREG/CR-6009
- ☆ PNNL Analyses
  - ☆ Core average and peak rod average at 35, 60, and 65 GWD/MTU
  - ☆ FRAPCON-3 Code w/ Missah release model
  - ☆ Best estimate approach
  - ☆ No operational transients addressed
  - ☆ Core inventories calculated to 75 GWD/MTU
- ☆ Base RG on PNNL analyses with some adjustments
- ☆ Balance uncertainty in gap fractions with other analysis conservatisms (e.g., peak burnup rod in peak power position)

Fuel Gap Fractions

	DG-1081	LOCA	Non-LOCA		RIA
Burnup, GWD/MTU=	0-62	0-62	0-40	40-62	0-62
I-131	0.12	0.05	0.05	0.08	0.1
Kr-85	0.15	0.05	0.06	0.1	0.3
Other NG	0.1	0.05	0.03	0.05	0.1
Other Halogen	0.1	0.05	0.03	0.05	0.1
Alkali Metals	0.1	0.05	0.08	0.12	0.0

- ☆ Gap fraction associated with peak rod burnup in core is used with rod inventory adjusted for maximum radial peaking factor
- ☆ Optional for accidents other than FHA, RIA, LOCA: if location of damaged fuel in core can be projected with reasonable certainty, can use assembly-specific gap fraction and radial peaking factors (not less than 1.0) on assembly-by-assembly basis

cc: Mr. Ralph Beedle
Senior Vice President
and Chief Nuclear Officer
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Mr. Alex Marion, Director
Programs
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Mr. David Modeen, Director
Engineering
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Mr. Anthony Pietrangelo, Director
Licensing
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Mr. Jim Davis, Director
Operations
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Mr. Robert R. Campbell, President
Nuclear HVAC Utilities Group
Tennessee Valley Authority
1101 Market Street, LP4J-C
Chattanooga, TN 37402-2801

Ms. Lynnette Hendricks, Director
Plant Support
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Mr. Kurt Cozens
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Mr. Dennis Adams
Nuclear HVAC Utilities Group
ComEd
1400 Opus Place
Downers Grove, IL 60515

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May 25, 2000

MEMORANDUM TO: Cynthia A. Carpenter, Chief
Generic Issues, Environmental, Financial &
Rulemaking Branch
Division of Regulatory Improvement Programs, NRR

FROM: Joseph L. Birmingham, Project Manager /RA Signed by JBirmingham/
Generic Issues, Environmental, Financial &
Rulemaking Branch
Division of Regulatory Improvement Programs, NRR

SUBJECT: MEETING WITH NUCLEAR ENERGY INSTITUTE (NEI) AND INDUSTRY REPRESENTATIVES CONCERNING CONTROL ROOM HABITABILITY AT OPERATING REACTORS

DATE & TIME: June 13-15, 8:00 a.m - 5:00 p.m.
Systems Group - 8:00 a.m - 12:00 p.m. June 13
Design Basis/Licensing Group - 1:00 - 5:00 p.m. June 14
Analysis Group - 8:00 a.m - 12:00 p.m. June 14 and 15

LOCATION: U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852
Room O-9B4 June 13 and 14
Room O-6B4 June 15

PURPOSE: NRC staff to continue discussions on NEI 99-03, "Control Room Habitability Assessment Guidance" and associated issues with the Design Basis/Licensing, Analyses and Systems Subgroups of the NEI Task Force on Control Room Habitability. Proposed agenda attached.

*PARTICIPANTS:

NRC	NEI
M. Reinhart	K. Cozens, et al.
J. Hayes, et al.	

Project No. 689
cc: See list
Attachment: As stated

Accession No.: ML003718568
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 25, 2000

MEMORANDUM TO: Cynthia A. Carpenter, Chief
Generic Issues, Environmental, Financial &
Rulemaking Branch
Division of Regulatory Improvement Programs, NRR

FROM: Joseph L. Birmingham, Project Manager
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cc: Mr. Ralph Beedle
Senior Vice President
and Chief Nuclear Officer
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Mr. Alex Marion, Director
Programs
Nuclear Energy Institute
Suite 400
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Washington, DC 20006-3708

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Engineering
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Mr. Anthony Pietrangelo, Director
Licensing
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Mr. H. A. Sepp, Manager
Regulatory and Licensing Engineering
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, Pennsylvania 15230

Mr. Jim Davis, Director
Operations
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Ms. Lynnette Hendricks, Director
Plant Support
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Mr. Charles B. Brinkman, Director
Washington Operations
ABB-Combustion Engineering, Inc.
12300 Twinbrook Parkway, Suite 330
Rockville, Maryland 20852

Mr. Kurt Cozens
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Proposed Agenda
NEI Task Force on Control Room Habitability Subgroup Meetings
June 13-15, 2000, Rooms O-9B4 and O-6B4

Introductory Remarks - Room O-9B4		M. Reinhart K. Cozens
June 13-14, 8AM-12PM	Systems Subgroup only - Room O-9B4	
June 14, 1PM-5PM	Design Basis/Licensing Subgroup only - Room O-9B4	
June 15, 8PM-12PM	Analysis Subgroup only - Room O-6B4	
Summary		M. Reinhart K. Cozens

Attachment