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INDIANA MICHIGAN POWER

June 12, 2000

C0600-13
10 CFR 50.90

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Units 1 and 2
LICENSE AMENDMENT REQUEST FOR CONTROL ROOM HABITABILITY AND GENERIC LETTER 99-02 REQUIREMENTS

Reference: Letter from J. F. Stang, Jr., Nuclear Regulatory Commission, to R. P. Powers, Indiana Michigan Power Company, "D.C. Cook Units 1 and 2 - Control Room Ventilation System Review", dated October 28, 1998

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend the Facility Operating Licenses, DPR-58 and DPR-74. Additionally, I&M proposes to use the methodology and the alternative source term (AST) in 10 CFR 50.67 and described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," and draft Regulatory Guide 1081, "Alternative Radiological Source Terms for Evaluating the Radiological Consequences of Design Basis Accidents at Boiling and Pressurized Water Reactors."

Implementing the AST of 10 CFR 50.67 results in a new acceptance criterion for 10 CFR 50, Appendix A, General Design Criterion (GDC) 19, of 5 rem total effective dose equivalent (TEDE). I&M has determined that use of the revised analysis assumptions, methodology, and acceptance criterion represents an unreviewed safety question. The NRC requires, in 10 CFR 50.67, a license

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amendment request to implement the AST as a replacement for the Technical Information Document (TID)-14844 source term. Therefore, NRC review and approval is required prior to implementation. I&M also proposes to amend Appendix A, "Technical Specifications" (T/S), to DPR-58 and DPR-74. The proposed changes are:

- separating the pressurization/filtration function requirements from the temperature control function in the control room emergency ventilation system (CREVS);
- adding a limiting condition for operation for the control room envelope/pressure boundary;
- modifying the requirements of the CREVS to add a new action, modify a surveillance requirement, and add a definition in the bases for the control room envelope/pressure boundary;
- expanding the applicability requirements and associated actions for CREVS to include, "during the movement of irradiated fuel assemblies";
- modifying action requirements for CREVS to address recently installed redundant dampers;
- incorporating the recommendations described in Generic Letter (GL) 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal", into the T/S for the CREVS, engineered safety features ventilation system (ESFVS), and storage pool ventilation system (SPVS);
- reducing the allowable pressure drop across the high-efficiency particulate air filter/charcoal adsorber units for the CREVS, ESFVS, and SPVS; and
- making editorial changes to the page format.

In response to the requested actions in GL 99-02, I&M is providing the current requirements for the laboratory testing of charcoal samples for the CREVS, ESFVS, and the SPVS.

Attachment 1 provides a detailed description and safety analysis to support the proposed changes. Attachments 2A and 2B provide marked up T/S pages for Unit 1 and Unit 2, respectively. Attachments 3A and 3B provide the proposed T/S pages with the changes incorporated for Unit 1 and Unit 2, respectively. Attachment 4 describes the evaluation performed in accordance with 10 CFR 50.92(c), which concludes that no significant hazard is involved. This conclusion is consistent with the NRC's position regarding "no significant

hazard" as discussed in the Federal Register dated December 23, 1999. Attachment 5 provides the environmental assessment.

In the referenced letter, the NRC requested I&M to submit a revised control room dose analysis demonstrating the conformance of the control room ventilation system with GDC-19 prior to the restart of either unit. Attachment 6 provides this analysis. Attachment 7 contains a listing of the design and operational issues for the CREVS identified by the NRC and Argonne National Laboratory and the resolution of those issues. Attachment 8 contains an evaluation of the post-loss-of-coolant accident (LOCA) equipment qualification dose based on the use of the AST. Attachment 9 provides a listing of the regulatory commitments contained in this request.

Since the analysis in Attachment 6 uses a new methodology, which requires prior approval, an additional control room dose analysis was performed using currently licensed methodology for CNP. Based on the guidance provided in Revision 1 of GL 91-18, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," I&M has determined that all safety-related structures, systems, and components required for safe shutdown or accident mitigation are operable. Condition Reports 99-29181 and 99-29182 were initiated to document these conclusions. The results verify the control room can meet the requirements of GDC-19 with compensatory actions per GL 91-18. The NRC staff reviewed the operability analyses as part of the closeout of the CNP Restart Action Matrix established in accordance with NRC Manual Chapter 0350.

This submittal applies the AST to the control room dose analysis methodology. Full implementation of the AST will be complete after submittal and NRC approval of the revised offsite dose methodology. The submittal requesting NRC approval of the offsite methodology will be provided to the NRC in a separate request.

I&M will continue to follow the industry standards applicable to control room habitability issues and stay appraised of any Nuclear Energy Institute guidance for assessing areas that can affect control room habitability and NRC operational concerns.

No previous submittals affect T/S pages that are submitted in this request. If any future submittals affect these T/S pages, then I&M will coordinate changes to the pages with the NRC Project Manager to ensure proper T/S page control when the associated license amendment requests are approved.

Copies of this letter and its attachments are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91.

Should you have any questions, please contact Mr. Robert C. Godley, Director of Regulatory Affairs, at (616) 466-2698.

Sincerely,

R. P. Powers
Vice President

\dms

Attachments

c: J. E. Dyer
MDEQ - DW & RPD
NRC Resident Inspector
R. Whale

AFFIRMATION

I, Robert P. Powers, being duly sworn, state that I am Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this Request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

Robert P. Powers
Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 12th DAY OF June, 2000

Notary Public

My Commission Expires _____

PATRICIA A. EDDIE
NOTARY PUBLIC - BERRIEN CO. MICH
MY COMMISSION EXPIRES
NOVEMBER - 5 - 2000

ATTACHMENT 1 TO C0600-13

DESCRIPTION AND SAFETY ANALYSIS FOR THE PROPOSED CHANGES

A. Summary of Proposed Changes

Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend the Facility Operating Licenses, DPR-58 and DPR-74. I&M proposes to use the methodology and the alternative source term (AST) described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," and draft Regulatory Guide (RG) 1081, "Alternative Radiological Source Terms for Evaluating the Radiological Consequences of Design Basis Accidents at Boiling and Pressurized Water Reactors."

I&M also proposes to amend Appendix A, "Technical Specifications" (T/S), to DPR-58 and DPR-74. The proposed changes are:

- separating the pressurization/filtration function requirements from the temperature control function in the control room emergency ventilation system (CREVS);
- adding a limiting condition for operation for the control room envelope/pressure boundary;
- modifying the requirements of the CREVS to add a new action, modify a surveillance requirement, and add a definition in the bases for the control room envelope/pressure boundary;
- expanding the applicability requirements and associated actions for CREVS to include, "during the movement of irradiated fuel assemblies";
- modifying action requirements for CREVS to address recently installed redundant dampers;
- incorporating the recommendations described in Generic Letter (GL) 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal" into the T/S for the CREVS, engineered safety features ventilation system (ESFVS), and storage pool ventilation system (SPVS); and
- reducing the allowable pressure drop across the high-efficiency particulate air (HEPA) filter/charcoal adsorber units for the CREVS, ESFVS, and SPVS;
- making editorial changes to the page format.

In response to the requested actions in GL 99-02, I&M is providing the current requirements for the laboratory testing of charcoal samples for the CREVS, ESFVS, and the SPVS.

B. Proposal to Implement the Alternative Source Term

Description of the Current Requirements

General Design Criterion (GDC)-19 limits radiological dose to control room personnel to fewer than or equal to 5 rem whole body, or its equivalent. The NRC has defined "or equivalent" as meaning 30 rem to the thyroid or 30 rem to the skin in NUREG-0800, "Standard Review Plan," Section 6.4. The radiological consequences analysis described in NUREG-0800 for a loss-of-coolant accident (LOCA) is based on the methodologies and assumptions derived from the Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" and other early guidance. The CNP control room habitability analysis was performed using this guidance to set the operating limits for the CREVS in order to ensure the requirements of GDC-19 were met. The dose consequences to personnel in the control room following the assumed accident are provided in Updated Final Safety Analysis Report (UFSAR), Section 14.3.5, "Environmental Consequences of a Loss-of-Coolant Accident."

The fission product release from the reactor core into the containment is the source term. It is characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release from the reactor core. The source term is used to evaluate the radiological consequences of design basis accidents in showing compliance with regulations.

Bases for the Current Requirements

CNP is licensed, in part, on the basis of safety analyses that used fission product release assumptions presented in TID-14844, which was published in March 1962. The control room dose assessment involves modeling of the radiological source term, the atmospheric transport of airborne activity, and the protection features of the CREVS. During an accident, the operators are exposed to radiation from an external cloud and radiation from activity buildup within the control room. The control room dose assessment models a LOCA using the TID-14844 source term, which assumes that 100% of the core noble gases and 50% of the core iodines are released instantaneously to the containment building. Of the iodine released to the containment building, 50% is assumed to plate out instantaneously on the containment building surfaces. Releases from the containment building are assumed to occur over the next 30-day period.

The thyroid dose is modeled using the iodine protection factor methodology given in the Murphy-Campe paper (K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting GDC-19," 13th AEC Air Cleaning Conference, August 1974). This model expresses the concentration of radioiodines in the control room as a fixed fraction of the concentration at the control room outdoor air intakes. The dose to personnel in the control room is calculated as a function of filtered intake and unfiltered inleakage. Filtered intake is that amount drawn by a pressurization fan through the emergency intake damper.

Unfiltered inleakage is a combination of the measured leakage past the normal intake damper, and a penalty accounting for opening and closing doors. The combination of filtered intake and unfiltered inleakage that results in a 30-rem thyroid dose is the most limiting condition. The GDC-19 limit is represented by an equation in Section 14.3.5 of the Unit 1 UFSAR showing makeup air flow as a function of unfiltered inleakage.

The NRC-approved methods for calculating the accident doses are described in RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." They were developed to be consistent with the TID-14844 source term and the whole body and thyroid dose acceptance criteria.

Need for Revision of the Requirements

I&M has determined that the limits of GDC-19 would be exceeded for a postulated design basis LOCA based on the revised values for the atmospheric dispersion factor (χ/Q) used for a building release from the containment building to the control room intake. The current analysis assumed the release from the containment building surface as the limiting value for χ/Q. Current industry standards indicate that event-specific release points should be evaluated to determine the most limiting release point. I&M evaluated these release points along with other input parameters that had not been considered previously. It was determined that additional margin would be needed to address use of specific release paths, additional input assumptions, and other outstanding control room habitability issues. I&M has reviewed recent NRC rulemaking and industry practice and has decided to adopt the AST. NUREG-1465 presents an AST for regulatory application for light-water reactors. The AST is described in terms of radionuclide composition and magnitude, physical and chemical form, and timing of release. Draft RG-1081 provides information to licensees regarding implementation of NUREG-1465. I&M determined that use of the AST would provide an acceptable margin to the revised GDC-19 limits.

10 CFR 50.67 requires licensees to request a license amendment to implement the AST as a replacement for the TID-14844 source term.

Description of the Proposed Changes

I&M proposes to use the AST described in 10 CFR 50.67 in a revised control room dose analysis and establish a new acceptance criterion for GDC-19 of 5 rem total effective dose equivalent (TEDE). The AST modeling results in several major departures from the assumptions used in the existing LOCA dose analysis reported in the UFSAR. Where TID-14844 addresses three categories of radionuclides, the AST categorizes the accident release into eight groups on the basis of similarity in chemical behavior. Instead of assuming instantaneous release of activity to the containment, the AST assumes the release of activity from the core occurs in phases over a 1.8-hour period, with the onset of major core damage occurring after 30 minutes. Where

TID-14844 assumes the radioiodine being primarily elemental, the AST assumes the radioiodine to be predominantly cesium iodide (CsI), an aerosol that is more amenable to mitigation mechanisms.

I&M also proposes to change the iodine protection factor methodology from that described in the Murphy-Campe paper to a discrete model of the control room volume. The Murphy-Campe paper expresses the concentration of the radioiodines in the control room as a fixed fraction of the concentration at the control room intake. The discrete volume model tracks radionuclide introduction to and removal from the control room volume as a function of time.

Bases for the Proposed Changes

Significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. Many of these insights developed out of the major research efforts started by the NRC and the nuclear industry after the accident at Three Mile Island. In 1995, the NRC published NUREG-1465, which used this research to provide more realistic estimates of the accident source term that could be applied to the design of light water reactors. To implement the methodology addressed in NUREG-1465, draft RG-1081 recommends licensees analyze the radiological consequences of design basis accidents. The dose analysis must address events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products postulated for dose purposes only. The dose consequences of events other than a LOCA are evaluated to address the different structures, systems, and components available to mitigate the dose consequences, since they may not be the same for all events. I&M has reanalyzed LOCA and non-LOCA events that could be limiting for control room dose using the AST and has included this report in Attachment 6.

I&M is proposing the AST methodology change for the revised control room dose analysis, as supported by the attached Westinghouse Electric Company report, "Revised Final Licensing Report for the Radiological Consequences of Accidents Using NUREG-1465 Source Term Methodology," in Attachment 6. The report is included in the first section of the attachment and the verified inputs assumed in the analysis are identified in the second section of the attachment, such as revised inleakage values due to plant modifications and reanalyzed points of origin for the χ/Q. The reanalyzed χ/Q release points include the containment vent, the power-operated relief valves (PORVs) located east and west of containment, the surface of the containment building, and the vent path on the Refueling Water Storage Tank (RWST). The second section contains an unverified input parameter not used in the control room dose analysis. The specific unverified input parameter, χ/Q for the releases from the containment building surface to the site boundary and low population zone, is not required for the control room dose analysis, only the offsite dose analysis. The assumed unfiltered inleakage to the control room reflects the recent tracer gas test results for control room inleakage. Both trains of CREVS pressurization fans are

assumed to start on each event that initiates a safety injection (SI) signal and manual operator action shuts down the second pressurization fan within two hours of the initiation of the event. The charcoal filters are assumed to remove 95% of the elemental iodine from the air directed through the beds when operating with a single pressurization fan. During the operation of two pressurization fans, the charcoal beds are assumed to remove 80% of iodine contained in the makeup air. The analysis considered 2 hours from the start of an event, as the limiting time for the operators to turn off the second pressurization fan. A removal efficiency of 98% of the particulate iodine is assumed for the HEPA filters. This assumption has been validated as part of the recent revisions to the Emergency Operating Procedures. The control room analysis models the control room as a discrete volume instead of using the iodine protection factor methodology.

The revised control room dose analysis considers an emergency core cooling system (ECCS) leak rate equivalent to 0.2 gallons per minute (gpm) of unfiltered leakage released to the atmosphere via the unit vent. The release of airborne activity from the ECCS leakage is assumed to occur during the recirculation phase for a design basis accident. A procedure is performed on an 18 month basis, which summarizes the leakage from the filtered and unfiltered areas of the recirculation flow path and any leakage back to the RWST to determine if the total effective recirculation leakage is within the limit.

The new dose analysis calculates the dose consequences to personnel in the control room for the following events: large break LOCA (LBLOCA), small break LOCA (SBLOCA), steam generator tube rupture (SGTR), locked rotor accident, rod ejection accident, fuel handling accident (FHA), main steam line break (SLB), gas decay tank (GDT) and volume control tank (VCT) rupture, and loss-of-offsite power (LOOP). The analysis demonstrates there is adequate radiation protection provided to permit access to, and occupancy of, the control room under the accident conditions evaluated without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. A summary of each accident analysis follows.

LBLOCA

In the LBLOCA analysis, all activity released from the fuel is assumed to be in the containment atmosphere until removed by sprays or the ice condenser, or from radioactive decay or leakage from the containment. The activity contained in the containment sump is recirculated through the ECCS piping during the recirculation phase. The analysis considers the equivalent of 0.2 gpm unfiltered ECCS leakage during the recirculation phase and applies an airborne fraction of 10^{-4} for the iodine contained in the ECCS leakage.

The value for the airborne fraction is an assumption allowed in the current analysis and referenced in the UFSAR, section 14.3.5, which states:

"The volatility of iodine from a simulated recirculation loop solution has been experimentally investigated. A solution including boric acid and sodium hydroxide was "spiked" with molecular

iodine and then evaporated to dryness at 200°F in a flowing air stream. The vapor generated by the evaporation process included iodine entrainment measured to be less than 10^{-4} of the iodine inventory in the original solution."

Justification is provided in a theoretical examination by A. E. J. Eggleton, titled "A Theoretical Examination of Iodine-water Partition Coefficients," published in 1967. This shows that a partitioning factor of less than 10^{-8} can be expected. The analysis also assumes a passive failure in the ECCS recirculation system 24 hours into the event, resulting in 50 gpm of leakage for one half-hour.

The LBLOCA analysis demonstrates a dose of fewer than 5.0 rem TEDE for personnel in the control room for the duration of the accident, which is within the requirements of 10 CFR 50.67.

SBLOCA

The SBLOCA event is not included in the analysis for the control room dose in the current licensing basis for CNP. This event was considered to be bounded by the LBLOCA event since spray actuation will occur for small breaks for typical ice condenser plants. However, there is a possibility that the operators could terminate containment spray for the SBLOCA events. Therefore, an analysis was done without any credit for iodine or particulate removal by the containment spray system.

To evaluate the SBLOCA, it is assumed that the break is small enough that the containment spray system is not actuated by high containment pressure. The core experiences some cladding damage such that the fission product gap activity of damaged fuel rods is released. The analysis conservatively assumes that the gap activity of all rods is released. The activity released to the containment is assumed to be released to the environment with the containment leaking at its design rate. There is also a release path through the steam generators (SG) totaling 1 gpm (T/S primary-to-secondary leakage limit) until the primary system pressure equalizes with the secondary system pressure.

The SBLOCA analysis demonstrates a dose of fewer than 5.0 rem TEDE for personnel in the control room for the duration of the accident, which is within the requirements of 10 CFR 50.67.

SGTR

For the SGTR event, the major hazard is the radiological consequences resulting from the transfer of radioactive reactor coolant to the secondary side of the ruptured SG and subsequent release of radioactive material to the atmosphere. The accident analyzed is the double-ended rupture of a single SG tube, resulting in the release of steam to the atmosphere via the SG PORVs and/or main steam safety valves (MSSVs). Following the SI actuation, it is assumed that the reactor coolant system (RCS) pressure stabilizes at the value where the SI and break flow

rates are equal. The equilibrium primary-to-secondary break flow is assumed to continue for 30 minutes after the initiation of the SGTR, at which time the operators complete the actions necessary to terminate the break flow and the steam releases from the ruptured SG.

It is recognized that the operators may not be able to terminate break flow within 30 minutes for all postulated SGTR events. The original methodology used for the SGTR analysis assumed a steady state break flow rate from the primary to secondary systems. A separate analysis has been performed using the licensed Westinghouse transient analysis code. The analysis used actual operator response times from simulator scenarios. The operator actions credited are identification and isolation of the ruptured SG, cooldown of the RCS by dumping steam from the intact SGs, depressurization of the RCS using the pressurizer PORV, and subsequent termination of the SI. The analysis demonstrates that although break flow may not be terminated within 30 minutes, the amount of radioactive reactor coolant flashing to steam and released to the atmosphere calculated with the assumption of a constant break flow rate for 30 minutes is more limiting than using the operator response times. The constant break flow rate is based on steady-state RCS and SG operating pressures prior to the reactor trip and the equilibrium RCS pressure considering ECCS injection and outgoing break flow after the reactor trip. Accordingly, I&M will maintain the original, more conservative analytical basis for CNP.

After the termination of break flow to the ruptured SG, the only steam released is from the intact SGs in order to dissipate the core decay heat and to cool the plant down to the residual heat removal (RHR) system operating conditions. The plant cool down to RHR operating conditions is not assumed to be completed until 30 days after initiation of the SGTR. The steam releases are terminated at that time.

The SGTR is analyzed considering both a pre-accident and accident-initiated iodine spike. The iodine activity concentration of the secondary coolant at the time the accident occurs is assumed to be equivalent to the T/S limit of 0.1 μCi/gm of DE I-131. For the pre-accident iodine spike case, it is assumed that a reactor transient has occurred prior to the SGTR and has raised the RCS iodine concentration to 60 times the T/S limit of 1.0 μCi/gm of dose equivalent (DE) I-131. For the accident-initiated iodine spike case, the reactor trip associated with the SGTR creates an iodine spike in the RCS that increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-cladding gap. Based on having 12% of the iodine in the fuel-cladding gap, the gap inventory would be depleted in six hours and the spike is terminated at that time.

The SGTR analysis demonstrates a dose of fewer than 5.0 rem TEDE for personnel in the control room for the duration of the accident, which is within the requirements of 10 CFR 50.67.

Locked Rotor Accident

For the locked rotor event, an instantaneous seizure of a reactor coolant pump rotor is assumed to occur, which rapidly reduces flow through the affected reactor coolant loop. In the locked rotor event analysis described in Attachment 6, no cladding failure is postulated because the peak cladding temperature (PCT) would not exceed 2700°F. The licensing basis analysis for a Unit 2 locked rotor event relies on the departure from nucleate boiling (DNB) ratio as a cladding failure criterion. This was established in the NRC Safety Evaluation Report for T/S changes associated with the Unit 2 Cycle 8 core reload (letter from T. G. Colburn, NRC, to M. P. Alexich, AEP, "Amendment Nos. 148 and 134 to Facility Operating License Nos. DRP-58 and DRP-74"). An analysis has been completed which demonstrates that no rods would experience DNB for a locked rotor event during Unit 2 fuel cycle 12. As a result, the dose consequences described in Attachment 6 are valid for Unit 2 during the current fuel cycle.

I&M will continue to conduct cycle-specific analyses of Unit 2 locked rotor events as needed to demonstrate that the event would not result in control room doses that exceed the 5 rem TEDE limits of 10 CFR 50.67. I&M will provide a supplement to this amendment request addressing the locked rotor event for Unit 1 prior to the unit entering Mode 2.

In the locked rotor event, there is a release path through the SGs (T/S primary-to-secondary leakage) until the primary system pressure equalizes with the secondary system pressure. A portion of this radioactive material is released to the outside atmosphere through either the SG PORVs or the MSSVs. The iodine activity concentration of the secondary coolant at the time the accident occurs is assumed to be equivalent to the T/S limit of 0.1 μCi/gm of DE I-131. In addition, iodine activity is contained in the secondary coolant prior to the accident and some of this is released to the atmosphere as a result of steaming from the SGs following the accident.

The control room dose would be bounded by the SLB and SGTR events. Therefore, a radiological consequences analysis is not necessary.

Rod Ejection Accident

For the rod ejection accident, it is assumed that a mechanical failure of a control rod drive mechanism (CRDM) pressure housing occurs, resulting in the ejection of a rod control cluster assembly and drive shaft. As a result of the accident, some fuel cladding damage and a small amount of fuel melt are assumed to occur. It is conservatively assumed that 15% of the fuel rods in the core are damaged. All of the fuel rod gap activity and one-half of the iodine activity from the melted fuel is assumed to be released to the RCS coolant.

There is also a release path through the SGs (T/S primary-to-secondary leakage) until the primary system pressure equalizes with the secondary system pressure. The iodine activity

concentration of the secondary coolant at the time the accident occurs is assumed to be equivalent to the T/S limit of 0.1 μCi/gm of DE I-131. A portion of this radioactive material is released to the outside atmosphere through the main condenser, SG PORVs, or MSSVs following the accident. The radioactive reactor coolant discharged to the containment from the rupture of the CRDM pressure housing is assumed to have a portion of its radioactive material released to the environment through containment leakage.

The rod ejection accident analysis demonstrates a dose of fewer than 5.0 rem TEDE for personnel in the control room for the duration of the accident, which is within the requirements of 10 CFR 50.67.

FHA

For an FHA, a fuel assembly is assumed to be dropped and damaged during refueling. The analysis of the accident is performed with assumptions selected so that the results are bounding for the accident occurring either inside containment or in the fuel handling building. The activity released from the damaged assembly is released to the outside atmosphere through either the containment purge system or the fuel handling building ventilation system. The control room χ/Q values for releases from the unit vent are chosen for use in this analysis since this bounds the 0 to 2 hour χ/Q values for releases from the containment building surface. Following an FHA, an assumption is made that the CREVS is placed in operation by manual operator action within 30 minutes.

It is assumed that all of the fuel rods of one fuel assembly are damaged to the extent that all of their gap activity is released. The fuel assembly radionuclide inventory is based on the assumption that the assembly had been operated at 1.65 times the core average power, with a 100-hour decay time. No credit is taken for the removal of iodine by SPVS filters, nor is credit taken for the isolation of release paths from containment by the purge line high radiation signal. The activity from the damaged fuel assembly is assumed to be released to the outside atmosphere over a two-hour period.

The FHA analysis demonstrates a dose of fewer than 5.0 rem TEDE for personnel in the control room for the duration of the accident, which is within the requirements of 10 CFR 50.67.

SLB

For the SLB, a complete severance of a main steam line outside containment is assumed. The faulted SG rapidly depressurizes and releases radioiodines initially contained in the secondary coolant and primary coolant (transferred via SG tube leaks) directly to the outside atmosphere. A portion of the iodine activity is released to the atmosphere through either the SG PORVs or the MSSVs. The SLB outside containment bounds any break inside containment, since the outside

break provides a means for direct release into the environment. The iodine activity concentration of the secondary coolant at the time the accident occurs is assumed to be equivalent to the T/S limit of 0.1 μCi/gm of DE I-131.

The SLB is analyzed considering both a pre-accident and accident-initiated iodine spike. For the pre-accident iodine spike case, it is assumed that a reactor transient has occurred prior to the SLB and has raised the RCS iodine concentration to 60 times the T/S limit of 1.0 μCi/gm of DE I-131. For the accident initiated iodine spike, the reactor trip associated with the SLB creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to the maximum equilibrium RCS T/S concentration of 1.0 μCi/gm of DE I-131. Based on 12% of the I-131 in the fuel-cladding gap, the gap inventory would be depleted in six hours and the spike would be terminated at that time.

The amount of primary-to-secondary SG tube leakage in the faulted SG is assumed to be 500 gallons per day (gpd). For the three intact SGs, the tube leakage is assumed to be 940 gpd (total), equal to the T/S limit for primary-to-secondary leakage. The leakage is assumed to occur for 30 days following the initiation of the SLB. The plant cooldown to RHR operating conditions is not assumed to be completed until 30 days after initiation of the SLB. The steam releases are terminated at that time.

No fuel failure is calculated to occur for the SLB event.

The SLB analysis demonstrates a dose of fewer than 5.0 rem TEDE for personnel in the control room for the duration of the accident, which is within the requirements of 10 CFR 50.67.

GDT and VCT Rupture

For the GDT rupture analysis, gas from one GDT is assumed to be released. For the VCT rupture analysis, gas from the VCT is assumed to be released, along with the noble gases from the reactor coolant letdown until the flow path from the RCS is isolated.

The inventory of the gases in the GDT is assumed to be the same as that of the entire RCS with 1% fuel defects. The inventory of gases in the VCT is based on continuous operation with 1% fuel defects without any purge of the VCT gas space. The inventory of iodine in the VCT is based on the reactor coolant activity at a pre-existing iodine spike level of 60 μCi/g DE I-131, with 90% of the iodine removed by the letdown demineralizer prior to entering the VCT. As a result of the accident, all of the noble gas in the tank and 1% of the iodine in the tank liquid are assumed to be released to the atmosphere over a period of five minutes.

The analysis assumes that the CREVS is in the normal mode of operation with a maximum of 1000 cfm of unfiltered makeup flow for the duration of the event. Actuation of the emergency

mode of operation would be initiated by a high radiation signal. Since CNP does not have a redundant radiation monitor, no credit is taken for the switchover to the emergency pressurization mode of the operation.

The GDT and VCT rupture analysis results demonstrate a dose of fewer than 5.0 rem TEDE for personnel in the control room for the duration of the accident, which is within the requirements of 10 CFR 50.67.

LOOP

For the LOOP, a loss of all AC power from the offsite sources to the unit and reserve auxiliary transformers is assumed. There is also a release path through the SGs (T/S primary-to-secondary leakage) until the primary system pressure equalizes with the secondary system pressure. The iodine activity concentration of the secondary coolant at the time the accident occurs is assumed to be equivalent to the T/S limit of 0.1 μCi/gm of DE I-131. A portion of this radioactive material is released to the outside atmosphere through the SG PORVs or MSSVs following the accident.

The LOOP is analyzed considering both a pre-accident and accident-initiated iodine spike. For the pre-accident iodine spike case, it is assumed that a reactor transient has occurred prior to the LOOP and has raised the RCS iodine concentration to 60 times the T/S limit of 1.0 μCi/gm of DE I-131. For the accident-initiated iodine spike case, the reactor trip associated with the LOOP creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater that the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-cladding gap. Based on having 12% of the I-131 in the fuel-cladding gap, the gap inventory would be depleted in six hours and the spike is terminated at that time.

The plant cooldown to RHR operating conditions is not assumed to be completed until 30 days after initiation of the LOOP and the steam releases are terminated at that time.

The LOOP event does not result in fuel cladding failures and an SI signal is not anticipated. Therefore, the CREVS is not assumed to switch from the normal operation mode to the emergency/pressurization mode of operation. The analysis assumes that the CREVS is in the normal mode of operation with a maximum of 1000 cfm of unfiltered makeup flow for the duration of the event.

The LOOP analysis demonstrates a dose of fewer than 5.0 rem TEDE for personnel in the control room for the duration of the accident, which is within the requirements of 10 CFR 50.67.

Summary of Accident Analysis Results

The calculated doses for each of the above events are provided in Attachment 6. The associated analyses demonstrate that the rod ejection accident is the dose-limiting event for personnel in the control room. The rod ejection accident is more dose-limiting due to the analysis assumption that containment spray operation would not occur, which would have removed the airborne particulates and elemental iodines. For each event, the analysis demonstrates a dose of fewer than 5.0 rem TEDE for personnel in the control room for the duration of the accident, which is within the requirements of 10 CFR 50.67.

The CREVS was not designed to meet the single failure criterion. Nevertheless, I&M has historically evaluated the impact of postulated single failures on the capability of the CREVS to mitigate doses to control room operators. I&M has performed modifications to eliminate the previously analyzed single failures. However, postulated single failures have been identified involving modifications to the emergency inlet dampers. Preliminary evaluation of the effects of these failures indicates that doses for two accidents (SBLOCA and Rod Ejection Accident) would be above the values given in Attachment 6, but well within the requirements of 10 CFR 50.67. I&M will finalize these evaluations and will provide a supplement to this submittal if the doses differ significantly from the values given in Attachment 6.

C. Proposed T/S Changes to T/S 3/4.7.5, "Control Room Emergency Ventilation System," and Associated Bases

Description of the Current Requirements

T/S 3/4.7.5.1 provides requirements for the CREVS, which includes both temperature control and filtration/pressurization functions. The CREVS is required to be operable in Modes 1 through 4. The required equipment for the filtration/pressurization function includes two independent pressurization fans and one charcoal adsorber and HEPA filter train. Surveillance Requirement 4.7.5.1.e.3 requires periodic demonstration that the control room is pressurized to the required pressure.

Bases for the Current Requirements

The operability of the CREVS ensures that the control room remains habitable for personnel during and following all credible accident conditions. The CREVS provides airborne radiological protection for the control room personnel for the most limiting design basis LOCA. This is demonstrated by the control room dose analysis presented in the UFSAR. The operability of this system, in conjunction with control room design provisions, is based on limiting the radiation exposure to personnel occupying the control room to fewer than or equal to 5 rem whole body, or its equivalent. This limitation is consistent with the requirements of GDC-19 of Appendix A to 10 CFR 50.

The existing temperature control function is based on environmental qualification limits for the equipment inside the control room envelope. The existing filtration/pressurization function is based on protecting control room personnel inside the control room pressure boundary/envelope from an uncontrolled release of radioactive material so that control room actions required following the release could be performed.

The surveillance to pressurize the control room verifies the integrity of the control room enclosure. The ability to maintain the control room at a positive pressure is periodically tested to verify proper operation of the CREVS and identify any additional unfiltered inleakage into the enclosure.

Need for Revision of the Requirements

GDC-19 requires a control room be provided for adequate radiation protection for personnel to permit access and occupancy under accident conditions without receiving excessive radiation exposures for the duration of the accident. An event that may occur during the movement of irradiated fuel is the design basis fuel handling accident. The event would require the protection available from the CREVS. However, the CREVS is not required to be operable in all cases by the current T/S when the movement of irradiated fuel could occur. Therefore, a revision to the applicability requirements is necessary.

I&M is installing an additional normal intake air damper in series with the existing normal intake damper and an additional emergency intake air damper in parallel with the existing emergency damper. These modifications have been completed on Unit 2 and will be completed on Unit 1 prior to the unit entering Mode 4. The CREVS was not designed to meet the single failure criterion. These modifications provide additional redundancy. The limiting condition for operation (LCO) and the action requirements should be revised to reflect installation of these two new dampers.

In performing the reanalysis of the control room dose, I&M assumed filtered makeup flow rate of fewer than or equal to 1000 cfm. In order to preserve this assumption, the value should be included as an acceptance criterion in the appropriate surveillance.

I&M is resolving issues related to control room temperature that may result in a future license amendment request. The control room cooling and filtration requirements are included in the same T/S. In order to simplify the future request for the cooling function, it is desirable to separate the two functions consistent with NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," Revision 1.

Control room surveys recommended that I&M define the control room envelope to include the control room, HVAC equipment room, and the plant process computer room. A new LCO, action, and surveillance requirement for maintaining and verifying the control room envelope is

required to maintain a consistent application of the volume defined as the control room envelope/pressure boundary. Additionally, the definition of the control room envelope and the allowance for normal ingress and egress of the area provides assurance that personnel are aware of the control room envelope/pressure boundary and able to recognize when the envelope is inadvertently breached in such a way that the accident analysis assumption is invalidated.

Description of the Proposed Changes

I&M proposes to include new applicability requirements and associated actions that would apply to the CREVS filtration/pressurization function during the movement of irradiated fuel assemblies. One proposed action addresses an inoperable pressurization train and requires action to restore the inoperable pressurization train within seven days or initiate and maintain operation of the remaining operable pressurization train in the pressurization/cleanup alignment. The second proposed action requires immediate suspension of all operations involving the movement of irradiated fuel assemblies if both pressurization trains, the filter unit, or the control room envelope/pressure boundary become inoperable.

I&M proposes to define a pressurization train in the Bases to include the pressurization fan, normal air intake damper, and emergency air intake damper. I&M also proposes to define a charcoal adsorber/HEPA filter unit in the Bases to include the prefilter, charcoal adsorber, HEPA filter, and filter housing. Corresponding changes are proposed for the LCO and actions to reflect the new definitions. I&M also proposes to add text to the Bases describing the applicability for the CREVS.

I&M proposes to add a new LCO that requires the control room envelope/pressure boundary to be able to maintain the required positive pressure for operability of the CREVS. If the control room envelope/pressure boundary is found inoperable, a proposed action requirement allows 24 hours to restore the control room envelope/pressure boundary. If the control room envelope/pressure boundary has not been restored within the allowed time, the unit must be shut down. I&M proposes to reflect the new term "control room envelope/pressure boundary" in the surveillance and to add a definition of the term to the Bases. I&M also proposes to revise the surveillance for verifying positive pressure. A test condition is added that limits makeup flow to fewer than or equal to 1,000 cfm.

I&M proposes to add a discussion in the Bases that provides allowance for normal ingress and egress to the control room under the administrative control of the person entering or exiting the area. For other openings, the allowance to open the control room envelope/pressure boundary intermittently under administrative control consists of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

I&M proposes to move the heating and cooling function of the CREVS from T/S 3/4.7.5.1 into a new, separate T/S, 3/4.7.5.2, for the control room air conditioning system, along with the associated action and surveillance requirement. The LCO, action, and surveillance requirement are relocated to the new T/S with no changes. I&M proposes to renumber the remaining conditions and action requirements in T/S 3/4.7.5.1.

Attachments 2A and 2B show the T/S pages marked to show the proposed changes.

Bases for the Proposed Changes

The accident analysis described in detail in Attachment 6 and summarized in the previous section demonstrates that the CREVS is needed for a fuel handling accident not previously credited. The proposed changes to the applicability requirements and associated actions provide protection for personnel in the control room for events that may occur during Modes 1, 2, 3, 4, or during the movement of irradiated fuel assemblies. The proposed changes for CREVS require the system to be operable to ensure that dose limits are maintained within the revised limits of GDC-19 during design basis accidents. The proposed actions ensure one train is operable and will perform its function. If the second pressurization train, the filter unit, or the control room envelope/pressure boundary is found inoperable during the movement of irradiated fuel assemblies, the action requires immediate suspension of the movement of irradiated fuel, which minimizes the possibility of a fuel handling accident.

The applicability change is consistent with NUREG-1431 because the CREVS is required to be operable to cope with the release from a fuel handling accident during the movement of irradiated fuel assemblies. It is also consistent with the recommendation in GL 83-37, "NUREG-0737 Technical Specifications (PWR)," and the NRC report titled "Brief Summary of September 15-19, 1986 Survey of the Control Room Ventilation System at D.C. Cook Units 1 and 2." These documents indicate that expansion of T/S applicability requirements should be considered to protect adequately the control room operators against the effects of an accidental release of radioactive gases.

In Modes 1, 2, 3, 4, and during movement of irradiated fuel assemblies, the CREVS must be operable to limit operator exposure during and following a design basis accident. In Modes 1, 2, 3, and 4, the system operability provides radiological protection to allow operators to take actions to mitigate the consequences of the accident. In conditions other than Modes 1, 2, 3, and 4, the only accident crediting the CREVS is the radioactive gas release from the fuel handling accident.

A toxic gas release has been analyzed and found not to be a concern for control room personnel. Hazardous materials stored within the CNP site boundary were examined for control room habitability concerns. It was concluded that the materials stored within the site boundary present no threat to plant safety, as release of these materials will not incapacitate control room personnel

or force a reactor shutdown. Current site procedures monitor the chemicals received by the plant to address the possibility of introducing a toxic gas source. Toxic chemical release from local industrial accidents or hazardous materials transportation accidents were also reviewed and determined to be insignificant. These concerns were discussed in the 1995 correspondence to the NRC titled, "Modification of Commitments Regarding Chlorine Detectors."

The proposed changes to the CREVS pressurization train terminology are a result of damper modifications being installed to enhance system performance. Adding new dampers provides additional redundancy and considers electrical separation. As such, the resulting suction side to the filtration unit can be considered a train. The proposed changes provide better alignment with NUREG-1431. The terminology differences are due to common components in the CNP CREVS design.

The proposed changes to define the control room envelope/pressure boundary and add a new LCO, action, and surveillance requirement provide assurance that the integrity of the control room envelope/pressure boundary is maintained and that the assumed inleakage rates of unfiltered air remain bounding. The proposed 24-hour allowed outage time and subsequent shutdown action are consistent with the requirements for an inoperable filter unit. The proposed action requirement is reasonable due to the low probability of the initiation of an accident requiring actuation of the CREVS occurring when the pressure boundary is inoperable.

The proposed change adds a discussion in the Bases that provides an allowance for normal ingress and egress to the control room for routine activities without entering the action statement. During the performance of the recent tracer gas test, normal ingress and egress into the control room was permitted through a single access. For other openings, the allowance to open the control room envelope/pressure boundary intermittently under administrative control allows routine repairs to be done without adversely affecting unit operations. The proposed 24-hour allowed outage time is consistent with Technical Specification Task Force 287, which is currently in NRC staff review.

The CREVS is designed to maintain a positive pressure in the control room envelope/pressure boundary with one train in the pressurization/cleanup mode with a makeup flow rate of fewer than or equal to 1,000 cfm. The proposed change to include this as a test condition preserves the accident analysis assumption.

The proposed change to separate the T/S into a pressurization/filtration T/S and heating/cooling T/S reflects the format used in NUREG-1431. The change to separate the two requirements, which includes renumbering of current requirements, is considered an administrative change.

D. Proposed Change to Incorporate Recommendations from GL 99-02 and Revise the HEPA Filter/Charcoal Adsorber Unit Pressure Drop for the CREVS, ESFVS, and SPVS

Description of the Current Requirements

T/S Surveillance requirements for CREVS define the testing requirements for the HEPA filters and charcoal adsorbers. The surveillances specify the sampling methods, requirements, and the testing standard to be used. The laboratory testing requirements demonstrate the charcoal adsorbers efficiencies are in accordance with the American National Standards Institute (ANSI) N510-1975 "Testing of Nuclear Air-Cleaning Systems." Surveillance requirements for laboratory analysis of the carbon samples verify a removal efficiency of greater than or equal to 90% for radioactive methyl iodide when tested at 130°C and a 95% relative humidity (RH).
The laboratory testing requirements for the charcoal adsorbers defined for the ESFVS and SPVS refer to ANSI N510-1980, "Testing of Nuclear Air-Cleaning Systems." ANSI N510-1980 endorses use of American Society for Testing and Materials (ASTM) D3803-1979 for laboratory testing. ASTM D3803-1979 is also explicitly included in the surveillance requirements. The carbon samples tested in the laboratory are required to demonstrate a removal efficiency of greater than or equal to 90% for radioactive methyl iodide when tested at 30°C and a 95% RH.

The T/S surveillance requirements for the CREVS, ESFVS, and SPVS include a pressure drop requirement of fewer than or equal to 6 inches water gauge across the combined charcoal adsorber and HEPA filter bank arrangement.

Bases for the Current Requirements

The sampling and laboratory testing methods for the charcoal in the CREVS are in accordance with ANSI N510-1975 to verify the components meet the efficiencies assumed in the accident analysis. It is assumed that testing the samples at a high temperature/high relative humidity represented the most severe conditions for the laboratory analysis of the carbon samples. The testing environment of 130°C and 95% RH is an option allowed by the standard and applied to the CREVS samples.

The ESFVS and SPVS surveillance requirements reference ANSI N510-1980 and ASTM D3803-1979 as the testing standards. Use of these testing standards was approved as a license amendment in 1987. I&M elected to use ANSI N510-1980 to comply more closely with the current industry practices for ESFVS and SPVS testing compared to use of the previous testing requirements of ANSI N510-1975. A testing environment of 30°C and 95% RH is applied to the laboratory analysis for the carbon samples, consistent with ANSI N510-1980 and with the safety evaluation report for the 1987 license amendment. The acceptance criterion for the samples is based on the iodine removal efficiency assumed in the accident analysis.

For the CREVS, ESFVS, and SPVS, the surveillance requirements for the HEPA filter/charcoal adsorber arrangements verify a pressure drop across the unit to identify a clogged or dirty filter needing replacement. The allowable value is the standard pressure drop value supplied by the manufacturer for a double HEPA filter arrangement.

Need for Revision of the Requirements

As described in GL 99-02, ANSI N510-1980 and earlier versions of the ANSI document provide the option to conduct the laboratory testing of the carbon samples at high temperatures and high RH. The NRC concluded that tests conducted at these higher temperatures do not provide consistent results of calculated charcoal efficiencies. Therefore, in GL 99-02, the NRC requested licensees to submit a license amendment either to change the charcoal testing standards to ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," or to submit an alternative methodology for NRC approval. The ASTM standard includes lower test temperatures to provide more consistent test results. I&M has reviewed ASTM D3803-1989 and proposes to change the testing standards to comply with the new standard. As a result, the T/S surveillance requirements need to be revised.

In GL 99-02, the NRC requests the licensee to provide the current requirements for the laboratory testing of charcoal samples for the CREVS, ESFVS, and the SPVS. For the CREVS, the testing protocol is ANSI N510-1975 at 130°C and 95% RH with a penetration depth of 10%. For the ESFVS and SPVS, the testing protocol is ANSI N510-1980 at 30°C and 95% RH with a penetration depth of 10%. The face velocity across each charcoal bed is fewer than or equal to 40 feet per minute. The charcoal beds are 2-inches thick with a residence time of greater than or equal to 0.25 seconds.

In GL 99-02, the NRC also recommended new terminology to express iodine removal efficiency, which includes a new safety factor of two. This safety factor provides additional safety margin. I&M does not use this terminology or safety factor in the T/S; therefore, the T/S need to be revised for consistency with the guidance.

The recent control room dose analysis assumes a charcoal adsorber efficiency of greater than or equal to 95% during normal operation of the CREVS system. This is not consistent with the 90% removal efficiency for the existing T/S surveillance requirements. Therefore, the acceptance criterion needs to be revised.

The filter unit pressure drop of 6 inches water gauge was designated for a double HEPA filter arrangement. However, the HEPA filter unit installed at CNP is a single HEPA filter unit. The 18-month surveillance requirements for the CREVS, ESFVS, and SPVS need to be revised to reduce the allowable pressure drop to fewer than or equal to 4 inches water gauge consistent with

the system design calculations. The system flow rate cannot be maintained using the existing T/S acceptance criterion of 6 inches water gauge.

Description of the Proposed Changes

I&M proposes to change the CREVS, ESFVS, and SPVS surveillance requirements to reference ASTM D3803-1989 as the standard for the laboratory testing requirements for the carbon samples. For the CREVS, the testing temperature is reduced from 130°C to 30°C, in accordance with the ASTM standard. The charcoal testing to the proposed requirements for the CREVS and ESFVS have been completed for Unit 2 and are scheduled to be completed prior to entry into Mode 4 for Unit 1. The SPVS charcoal testing has been completed.

I&M proposes to change the terminology used to express a charcoal removal efficiency and apply a safety factor. The ESFVS and SPVS would require the carbon samples removed from the charcoal adsorbers to show a penetration of fewer than or equal to 5% for radioactive methyl iodine, rather than a removal efficiency of 90%. The CREVS would require the carbon samples removed from the charcoal adsorbers to show a penetration of fewer than or equal to 1% for radioactive methyl iodine, rather than a removal efficiency of 90%. The acceptance criterion for the CREVS also reflects a change to reflect the recent control room dose analysis from 90% removal efficiency to 95% removal efficiency.

I&M proposes to change surveillance requirements for CREVS, ESFVS, and SPVS to verify a pressure drop of fewer than 4 inches water gauge rather than 6 inches water gauge for the combined HEPA filter and charcoal adsorber banks.

Attachments 2A and 2B show the T/S pages marked to show the proposed changes.

Bases for the Proposed Changes

The proposed laboratory testing standard, ASTM D3803-1989, provides a more accurate indication of degradation in the iodine removal capability of the charcoal adsorbers in the CREVS, ESFVS, and SPVS. The revised testing methodologies for performing the laboratory testing at the lower temperatures will provide more conservative values for calculated removal efficiency of the carbon samples due to more water being retained by the charcoal. As the temperature is increased, less moisture is retained by the charcoal, which allows more iodine to be removed. By updating the testing requirements to conform with the recommendations of GL 99-02, the testing environment for the carbon samples reflects the expected operating conditions of the charcoal during an accident.

In the recent control room dose analysis, an iodine removal efficiency of 95% is assumed for single-fan operation (normal system flow rate). Also assumed during the first two hours of the

accident is two-fan operation where an 80% charcoal adsorber removal efficiency is assumed. The 80% efficiency is calculated based on the increased face velocity across the charcoal bed and a 1% penetration allowance at the normal system flow rate. The 80% efficiency calculation also includes a safety factor of two. To ensure the accident analysis assumptions remain valid for both single- and two-fan operation, the surveillance requirement is revised to demonstrate a penetration of less than or equal to 1% when tested at the normal system flow rate. The acceptance criterion reflects the initial assumption of 99% removal efficiency used in the revised analysis. Using a safety factor of two would require a minimum removal efficiency of 98% for single-fan operation. Additional margin is gained for single-fan operation by using an acceptance criterion above that assumed in the analysis (98% vs. 95%).

The proposed change for the pressure drop acceptance criterion provides assurance that the system is able to maintain the design system flow rate. Calculations demonstrate that the 4 inches water gauge value is appropriate.

E. Proposed Administrative Changes

I&M is proposing certain format changes affecting each page that include minor differences in margins and text spacing due to variations in word processing and reprographic technologies. There are specific format changes affecting most of the revised pages. The following pages for Unit 1 have format changes included in their revision: 3/4 7-19, 3/4 7-20, 3/4 7-21, 3/4 7-24, 3/4 7-25, 3/4 9-14, and 3/4 9-15. For Unit 2, format changes are included in the following pages: 3/4 7-14, 3/4 7-15, 3/4 7-16, 3/4 7-18, 3/4 7-19, 3/4 9-13, and 3/4 9-14. These specific format changes include (1) the use of a different font, which also results in altered spacing of the text on the page and content for each line of text, (2) the use of horizontal bars to separate the footer and header from the body of the page, (3) the addition of numerical annotation (i.e., 3/4 and 3/4.7 in the header text lines), (4) the removal of underlining from the two lines of header text, (5) the reversal of sequence and deletion of a blank line between the two lines of header text, (6) the removal of spaces immediately preceding and following the hyphen in the footer text, "COOK NUCLEAR PLANT-UNIT," (7) the addition of the word "Page" prior to the page number, and (8) the removal of the word "NO." following the word "AMENDMENT" and prior to the historical and current amendment numbers.

Typographical errors were corrected in the Unit 1 T/S on pages: (1) 3/4 7-22 to add (W.G.) as an abbreviation for water gauge, and (2) 3/4 7-25 to correct band to bank. For Unit 2 T/S, corrections were made to the following pages: (1) 3/4 7-15 to move part d to the next page, (2) 3/4 7-16a to add (W.G.) as an abbreviation for water gauge, and (3) B3/4 7-4a, to put the word emergency in lower case. The proposed index pages for Unit 1 and 2 separate T/S 3/4.7.5 into two specifications and removes the word emergency from the major heading for the CREVS.

These format and editorial changes on each page are administrative and do not result in any change in the actual requirements.

F. Discussion of Risk

The proposed change to replace the traditional TID-14844 source term with the AST does not change the actual accident sequence and progression. I&M has reanalyzed LOCA and non-LOCA events that could be the limiting condition for control room dose using the AST. The use of the AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents. The AST has no direct effect on the probability of the accident. The use of an AST alone cannot increase the core damage frequency or the large early release frequency.

The proposed change to include a new mode of applicability that would apply to the CREVS filtration/pressurization function during movement of irradiated fuel assemblies provides assurance that the CREVS will perform its function during postulated events when it is required. The proposed actions ensure one train is operable and will perform its function. If no train is operable, the action requires immediate suspension of the movement of irradiated fuel assemblies, which precludes occurrence of the fuel handling accident. There is no increase in risk because the CREVS system would be available to perform its function when required.

I&M evaluated the modification to install redundant dampers in accordance with 10 CFR 50.59. The modification reduces the failure probability of the control room ventilation system by more than a factor of fifty. The proposed changes to the LCOs and actions for the redundant dampers provide assurance that the reliability of the CREVS will increase compared to the current system design.

The proposed action requirement to allow 24 hours to restore the inoperable control room envelope/pressure boundary prior to initiating a plant shutdown is reasonable due to the low probability of an accident occurring during the period of time the pressure boundary could be inoperable.

The remaining proposed changes are administrative and have no impact on risk.

G. Schedule Requirements

I&M requests NRC approval in a timely manner. There is no specific request date for NRC approval. I&M has performed an operability assessment in accordance with GL 91-18, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," Revision 1, to ensure the control room meets the requirements of GDC-19. Administrative controls have been developed to support restart and subsequent operation. These controls would be discontinued, as appropriate, after the proposed changes are implemented.

ATTACHMENT 2A TO C0600-13

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MARKED TO SHOW PROPOSED CHANGES

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~~3/4.7.5~~ CONTROL ROOM VENTILATION SYSTEM

~~3/4.7.5~~CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.1 The control room emergency ventilation system (CREVS) shall be OPERABLE with:

~~a. Two independent heating and cooling systems,~~

~~b.~~a. Two independent pressurization ~~fans~~ trains,

~~c.~~b. One charcoal adsorber ~~and~~ /HEPA filter ~~train~~ unit, and

c. The control room envelope/pressure boundary able to maintain the required positive pressure.

APPLICABILITY: MODES 1, 2, 3, ~~and~~ 4, and during the movement of irradiated fuel assemblies.

ACTION:

MODES 1,2,3, and 4:

~~a. With one heating and cooling system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~b.~~a. With one pressurization ~~fan~~ train inoperable, restore the inoperable train ~~fan~~ to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~c.~~b. With the filter ~~train~~ unit inoperable, restore the filter ~~train~~ unit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the control room envelope/pressure boundary inoperable, restore the control room envelope to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

During the movement of irradiated fuel assemblies:

d. With one pressurization train inoperable, restore the inoperable pressurization train to OPERABLE status within 7 days, or initiate and maintain operation of the remaining OPERABLE train in the pressurization/cleanup alignment.

e. With any of the following (1) both pressurization trains inoperable; (2) the filter unit inoperable; or (3) the control room envelope/pressure boundary inoperable, immediately suspend all operations involving the movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.5.1 The control room emergency ventilation system shall be demonstrated OPERABLE:

a. ~~At least once per 12 hours by verifying that the control room air temperature is less than or equal to 95°F.~~ Deleted

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

2. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 1.0% ~~demonstrates a removal efficiency of ≥ 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, ~~ANSI N510-1975 (130~~30°C, 95% R.H.~~)~~. The carbon samples not obtained from test canisters shall be prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

d. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 1.0%~~demonstrates a removal efficiency of ≥ 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, ~~ANSI N510-1975 (130~~30°C, 95% R.H.~~)~~; or

2. Verifying within 31 days after removal that a laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 1.0%~~demonstrate a removal efficiency of ≥ 90%~~ for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, ~~ANSI N510-1975 (130~~30°C, 95% R.H.~~)~~; and the samples are prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

a) Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%, and

b) Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~6~~ 4 inches Water Gauge (W. G.) while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

2. a. Verifying that on a Safety Injection Signal from Unit 1, the system automatically operates in the pressurization/cleanup mode.

   b. Verifying that on a Safety Injection Signal from Unit 2, the system automatically operates in the pressurization/cleanup mode.

3. Verifying that the system maintains the control room envelope/pressure boundary at a positive pressure of greater than or equal to 1/16 inch W. G. relative to the outside atmosphere at a system flow rate of 6000 cfm plus or minus 10%, with a makeup air flow rate of ≤ 1000 cfm.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

CONTROL ROOM AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.2 The control room air conditioning system (CRACS) shall be OPERABLE with two heating and cooling systems.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one heating and cooling system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.2 The control room air conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is less than or equal to 95°F.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% ~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ~~ANSI N510-1980~~ ASTM D3803-1989 ~~(ASTM D-3803-1979~~, 30°C, 95% R.H.~~)~~. The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 25,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5%~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989 ~~ANSI N510-1980 (ASTM D-3803-1979,~~ 30°C, 95% R.H.~~)~~; or

2. Verifying within 31 days after removal that laboratory analyses of at least two carbon samples shows a penetration of less than or equal to 5%~~demonstrate a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D-3803-1979~~, 30°C, 95% R.H.~~)~~ and the samples are prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

SURVEILLANCE REQUIREMENTS (Continued)

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%:

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~6~~ 4 inches Water Gauge while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

2. Deleted.

3. Verifying that the standby fan starts automatically on a Containment Pressure--High-High Signal and directs its exhaust flow through the HEPA filters and charcoal adsorber banks on a Containment Pressure--High-High Signal.

e. After each complete or partial replacement of HEPA filter ~~band~~ bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% ~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D-3803-1979~~, 30°C, 95% R.H.~~)~~. The carbon samples not obtained from test canisters shall be prepared by either:

   (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 30,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

   1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5% ~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989 ~~ANSI N510-1980 (ASTM D-3803-1979~~, 30°C, 95% R.H.~~)~~; or

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that laboratory analyses of at least two carbon samples show a penetration of less than or equal to 5% ~~demonstrate a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D-3803-1979, 30°C, 95% R.H.)~~ and the samples are prepared by either:

   (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~or equal to 6~~ 4 inches Water Gauge while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

   2. Deleted.

   3. Verifying that on a high-radiation signal, the system automatically directs its exhaust flow through the charcoal adsorber banks and automatically shuts down the storage pool ventilation system supply fans.

   4. Verifying that the exhaust ventilation system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION AND CONTROL ROOM AIR CONDITIONING SYSTEMS

The OPERABILITY of the control room emergency ventilation system (CREVS) ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. In MODES 1-4, the CREVS provides radiological protection to allow operators to take the actions necessary to mitigate the consequences of a design basis accident. The CREVS is also required to be OPERABLE for operations involving the movement of irradiated fuel assemblies to provide protection from a fuel handling accident. The CREVS operation is not credited during the rupture of a waste gas tank or toxic gas release. The CREVS has two pressurization trains with each pressurization train consisting of a pressurization fan, normal intake air damper, and emergency intake air damper available to align and maintain flow to the control room. The charcoal adsorber/HEPA filter unit consists of the prefilter, charcoal adsorbers, HEPA filter, and filter housing. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to less than or equal to 5 rem Total Effective Dose Equivalent, TEDE ~~or less whole body, or its equivalent~~. This limitation is consistent with the requirements of General Design Criteria (GDC) 19 of Appendix "A", 10 CFR 50.

The control room envelope/pressure boundary consists of the control room, the control room HVAC equipment room, and the plant process computer room and shall be considered OPERABLE if these rooms can be maintained at a positive pressure of greater than or equal to 1/16 inch water gauge relative to the outside atmosphere. If the control room envelope/pressure boundary is inoperable in MODES 1, 2, 3, and 4, actions must be taken to restore an OPERABLE control room boundary within 24 hours. The control room boundary can be opened intermittently under administrative controls. For entry and exit through doors to the control room, the administrative control of opening the pressure boundary is performed by the person entering or exiting the area. For other openings of the control room envelope/pressure boundary, the administrative controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

The Unit 1 control room emergency ventilation system aligns and operates automatically on a Safety Injection (SI) Signal from either Unit 1 or Unit 2. Both pressurization fans start on the SI signal. Procedures direct realignment of the CREVS to single fan operation within two hours after receiving the SI signal. The automatic start from Unit 2 is normally only available when the Unit 2 ESF actuation system is active in modes 1 through 4 in Unit 2.

The control room air conditioning ~~ventilation~~ system (CRACS) normally maintains the control room at temperatures at which control room equipment is qualified for the life of the plant. Continued operation at the Technical Specification limit is permitted since the portion of time the temperature is likely to be elevated is small in comparison to the qualified life of the equipment at the limit.

3/4.7.6 ESF VENTILATION SYSTEM

The OPERABILITY of the ESF ventilation system ensures that adequate cooling is provided for ECCS equipment and that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations were assumed in the accident analysis.

The 1980 version of ANSI N510 is used as a testing guide. This standard, however, is intended to be rigorously applied only to systems which, unlike the ESF ventilation system, are designed to ANSI N509 standards. For the specific case of the air-aerosol mixing uniformity test required by ANSI N510 as a prerequisite to in-place leak testing of charcoal and HEPA filters, the air-aerosol uniform mixing test acceptance criteria were not rigorously met. For this reason, a statistical correction factor will be applied to applicable surveillance test results where required.

3/4.7.7 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, are based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3/4.7.8 HYDRAULIC SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

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3/4.7.5 CONTROL ROOM VENTILATION SYSTEMS

~~3/4.7.5~~ CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.1 The control room emergency ventilation system (CREVS) shall be OPERABLE with:

~~a. Two independent heating and cooling systems,~~

~~b.~~a. Two independent pressurization ~~fans~~trains,

~~c.~~b. The charcoal adsorber~~/ and~~ HEPA filter ~~train~~unit, and

c. The control room envelope/pressure boundary able to maintain the required positive pressure.

APPLICABILITY: MODES 1, 2, 3, ~~and~~ 4, and during the movement of irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3, and 4:

~~a. With one heating and cooling system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~b.~~a. With one pressurization ~~fan~~ train inoperable, restore the inoperable ~~fan~~ train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~c.~~b. With the filter ~~train~~ unit inoperable, restore the filter ~~train~~ unit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the control room envelope/pressure boundary inoperable, restore the control room envelope to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

During movement of irradiated fuel assemblies:

d. With one pressurization train inoperable, restore the inoperable pressurization train to OPERABLE status within 7 days, or initiate and maintain operation of the remaining OPERABLE train in the pressurization/cleanup alignment.

e. With any of the following (1) both pressurization trains inoperable; (2) the filter unit inoperable; or (3) the control room envelope/pressure boundary inoperable, immediately suspend all operations involving the movement of irradiated fuel assemblies

SURVEILLANCE REQUIREMENTS

4.7.5.1 The control room emergency ventilation system shall be demonstrated OPERABLE:

a. ~~At least once per 12 hours by verifying that the control room air temperature is less than or equal to 95°F.~~Deleted

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the system operates for at least 15 minutes.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

2. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 1.0% ~~demonstrates a removal efficiency of ≥ 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, ~~ANSI N510-1975 (130~~30°C, 95% R.H.~~)~~. The carbon samples not obtained from test canisters shall be prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

d. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 1.0% ~~demonstrates a removal efficiency of ≥ 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, ~~ANSI N510-1975 (130~~30°C, 95% R.H.~~)~~; or

2. Verifying within 31 days after removal that a laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 1.0% ~~demonstrate a removal efficiency of ≥ 90%~~ for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, ~~ANSI N510-1975 (130~~30°C, 95% R.H.~~)~~ and the samples are prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

a) Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%, and

b) Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~6~~ 4 inches Water Gauge (W.G.) while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

2. a. Verifying that on a Safety Injection Signal from Unit 1, the system automatically operates in the pressurization/cleanup mode.

   b. Verifying that on a Safety Injection Signal from Unit 2, the system automatically operates in the pressurization/cleanup mode.

3. Verifying that the system maintains the control room envelope/pressure boundary at a positive pressure of greater than or equal to 1/16 inch W. G. relative to the outside atmosphere at a system flow rate of 6000 cfm plus or minus 10% with a makeup air flow rate of ≤ 1000 cfm.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

CONTROL ROOM AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.2 The Control room air conditioning system (CRACS) shall be OPERABLE with two heating and cooling systems.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one heating and cooling system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.2 The control room air conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is less than or equal to 95°F.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% ~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D 3803-1979~~, 30°C, 95% R.H.~~)~~. The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 25,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5% ~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D 3803-1979~~, 30°C, 95% R.H.~~)~~; or

2. Verifying within 31 days after removal that laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 5% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989ANSI N510-1980~~(ASTM D3803-1979, 30°C, 95% R.H.)~~ and the samples are prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

SURVEILLANCE REQUIREMENTS (Continued)

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

2. Deleted.

3. Verifying that the standby fan starts automatically on a Containment Pressure--High-High Signal and directs its exhaust flow through the HEPA filters and charcoal adsorber banks on a Containment Pressure--High-High Signal. +

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

+ The provisions of Technical Specification 4.0.8 are applicable.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% ~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D-3803-1989~~ANSI N510-1980 (ASTM D-3803-1979~~, 30°C, 95% R.H.~~)~~. The carbon samples not obtained from test canisters shall be prepared by either:

   (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 30,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

   1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5% ~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D-3803-1989~~.ANSI N510-1980 (ASTM D 3803-1979~~, 30°C, 95%~~,~~ R.H.~~)~~

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that laboratory analyses of at least two carbon samples show a penetration of less than or equal to 5% ~~demonstrate a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the samples are tested in accordance with ASTM D-3803-1989~~ANSI N510-1980 (ASTM D-3803-1979~~, 30°C, 95% R.H.~~)~~ and the samples are prepared by either:

   (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~or equal to 6~~ 4 inches Water Gauge while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

   2. Deleted.

   3. Verifying that on a high-radiation signal, the system automatically directs its exhaust flow through the charcoal adsorber banks and automatically shuts down the storage pool ventilation system supply fans.

   4. Verifying that the exhaust ventilation system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the control room ~~EMERGENCY~~emergency ventilation system (CREVS) ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. In MODES 1-4, the CREVS provides radiological protection to allow operators to take the actions necessary to mitigate the consequences of a design basis accident. The CREVS is also required to be OPERABLE for operations involving the movement of irradiated fuel assemblies to provide protection from a fuel handling accident. The CREVS has two pressurization trains with each pressurization train consisting of a pressurization fan, normal intake air damper, and emergency intake air damper available to align and maintain flow to the control room. The charcoal adsorber/HEPA filter unit consists of the prefilter, charcoal adsorber, HEPA filter, and filter housing. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to less than or equal to 5 rem Total Effective Dose Equivalent, TEDE. ~~5 rem or less whole body, or its equivalent.~~ This limitation is consistent with the requirements of General Design Criteria (GDC) 19 of Appendix "A", 10 CFR 50.

The control room envelope/pressure boundary consists of the control room, the control room HVAC equipment room, and the plant process computer room and shall be considered OPERABLE if these rooms can be maintained at a positive pressure of greater than or equal to 1/16 inch water gauge relative to the outside atmosphere. If the control room envelope/pressure boundary is inoperable in MODES 1, 2, 3, and 4, actions must be taken to restore an OPERABLE control room boundary within 24 hours. The control room boundary can be opened intermittently under administrative controls. For entry and exit through doors to the control room, the administrative control of opening the pressure boundary is performed by the person entering or exiting the area. For other openings of the control room envelope/pressure boundary, the administrative controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

The Unit 2 control room emergency ventilation system aligns and operates automatically on a Safety Injection (SI) Signal from either Unit 1 or Unit 2. Both pressurization fans start on the SI signal. Procedures direct realignment of the CREVS to single fan operation within two hours after receiving the SI signal. The automatic start from Unit 1 is normally only available when the Unit 1 ESF actuation system is active in modes 1 through 4 in Unit 1.

The Limiting Condition for Operation requires two independent control room heating and cooling systems. Each cooling system requires a functional air handling unit and associated cooling water supply. Cooling water is provided from a chilled water unit. At the design maximum essential service water (ESW) supply temperature of 86°F, a chilled water unit will maintain the control room temperature below 95°F. Cooling water may also be supplied directly by ESW when ESW supply temperature is ≤ 65°F.

The control room ~~ventilation~~ air conditioning system (CRACS) normally maintains the control room at temperatures at which control room equipment is qualified for the life of the plant. Continued operation at the Technical Specification limit is permitted since the portion of time the temperature is likely to be elevated is small in comparison to the qualified life of the equipment at the limit.

Each control room cooling system can maintain control room temperature ≤ 102°F during accident conditions with the control room isolated. At control room temperatures of ≤ 102°F, vital control room equipment remains within its manufacturer's recommended operating temperature range.

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3/4.7.5 CONTROL ROOM VENTILATION SYSTEM

CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.1 The control room emergency ventilation system (CREVS) shall be OPERABLE with:

a. Two independent pressurization trains,

b. One charcoal adsorber/HEPA filter unit, and

c. The control room envelope/pressure boundary able to maintain the required positive pressure.

APPLICABILITY: MODES 1, 2, 3, 4, and during the movement of irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3, and 4:

a. With one pressurization train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the filter unit inoperable, restore the filter unit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the control room envelope/pressure boundary inoperable, restore the control room envelope to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

During the movement of irradiated fuel assemblies:

d. With one pressurization train inoperable, restore the inoperable pressurization train to OPERABLE status within 7 days, or initiate and maintain operation of the remaining OPERABLE train in the pressurization/cleanup alignment.

e. With any of the following (1) both pressurization trains inoperable; (2) the filter unit inoperable; or (3) the control room envelope/pressure boundary inoperable, immediately suspend all operations involving the movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.5.1 The control room emergency ventilation system shall be demonstrated OPERABLE:

a. Deleted

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

2. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 1.0% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. The carbon samples not obtained from test canisters shall be prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (continued)

d. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 1.0% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H; or

2. Verifying within 31 days after removal that a laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 1.0% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. and the samples are prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

a) Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%, and

b) Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 4 inches Water Gauge (W. G.) while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

2. a. Verifying that on a Safety Injection Signal from Unit 1, the system automatically operates in the pressurization/cleanup mode.

   b. Verifying that on a Safety Injection Signal from Unit 2, the system automatically operates in the pressurization/cleanup mode.

3. Verifying that the system maintains the control room envelope/pressure boundary at a positive pressure of greater than or equal to 1/16 inch W. ~~G.~~ relative to the outside atmosphere at a system flow rate of 6000 cfm plus or minus 10%, with a makeup air flow rate of ≤ 1000 cfm.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

CONTROL ROOM AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.2 The control room air conditioning system (CRACS) shall be OPERABLE with two heating and cooling systems.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one heating and cooling system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.2 The control room air conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is less than or equal to 95°F.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 25,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980 .

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5%for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H.; or

2. Verifying within 31 days after removal that laboratory analyses of at least two carbon samples shows a penetration of less than or equal to 5%for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. and the samples are prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

SURVEILLANCE REQUIREMENTS (Continued)

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%:

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 4 inches Water Gauge while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

2. Deleted.

3. Verifying that the standby fan starts automatically on a Containment Pressure--High-High Signal and directs its exhaust flow through the HEPA filters and charcoal adsorber banks on a Containment Pressure--High-High Signal.

e. After each complete or partial replacement of HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. The carbon samples not obtained from test canisters shall be prepared by either:

   (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 30,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980 .

c. After every 720 hours of charcoal adsorber operation by either:

   1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H; or

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that laboratory analyses of at least two carbon samples show a penetration of less than or equal to 5% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. and the samples are prepared by either:

   (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 4 inches Water Gauge while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

   2. Deleted.

   3. Verifying that on a high-radiation signal, the system automatically directs its exhaust flow through the charcoal adsorber banks and automatically shuts down the storage pool ventilation system supply fans.

   4. Verifying that the exhaust ventilation system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION AND CONTROL ROOM AIR CONDITIONING SYSTEMS

The OPERABILITY of the control room emergency ventilation system (CREVS) ensures that the control room will remain habitable for operations personnel during ~~and following all credible~~ accident conditions. In MODES 1-4, the CREVS provides radiological protection to allow operators to take the actions necessary to mitigate the consequences of a design basis accident. The CREVS is also required to be OPERABLE for operations involving the movement of irradiated fuel assemblies to provide protection from a fuel handling accident. The CREVS operation is not credited during the rupture of a waste gas tank or toxic gas release. The CREVS has two pressurization trains with each pressurization train consisting of a pressurization fan, normal intake air damper, and emergency intake air damper available to align and maintain flow to the control room. The charcoal adsorber/HEPA filter unit consists of the prefilter, charcoal adsorbers, HEPA filter, and filter housing. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to less than or equal to 5 rem total Effective Dose Equivalent, TEDE. The limitation is consistent with the requirements of General Design Criteria (GDC) 19 of Appendix "A", 10 CFR 50.

The control room envelope/pressure boundary consists of the control room, the control room HVAC equipment room, and the plant process computer room and shall be considered OPERABLE if these rooms can be maintained at a positive pressure of greater than or equal to 1/16 inch water gauge relative to the outside atmosphere. If the control room envelope/pressure boundary is inoperable in MODES 1, 2, 3, and 4, actions must be taken to restore an OPERABLE control room boundary within 24 hours. The control room boundary can be opened intermittently under administrative controls. For entry and exit through doors to the control room, the administrative control of opening the pressure boundary is performed by the person entering or exiting the area. For other openings of the control room envelope/pressure boundary, the administrative controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

The Unit 1 control room emergency ventilation system aligns and operates automatically on a Safety Injection (SI) Signal from either Unit 1 or Unit 2. Both pressurization fans start on the SI signal. Procedures direct realignment of the CREVS to single fan operation within two hours after receiving the SI signal. The automatic start from Unit 2 is normally only available when the Unit 2 ESF actuation system is active in modes 1 through 4 in Unit 2.

The control room air conditioning system (CRACS) normally maintains the control room at temperatures at which control room equipment is qualified for the life of the plant. Continued operation at the Technical Specification limit is permitted since the portion of time the temperature is likely to be elevated is small in comparison to the qualified life of the equipment at the limit.

3/4.7.6 ESF VENTILATION SYSTEM

The OPERABILITY of the ESF ventilation system ensures that adequate cooling is provided for ECCS equipment and that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations were assumed in the accident analysis.

The 1980 version of ANSI N510 is used as a testing guide. This standard, however, is intended to be rigorously applied only to systems which, unlike the ESF ventilation system, are designed to ANSI N509 standards. For the specific case of the air-aerosol mixing uniformity test required by ANSI N510 as a prerequisite to in-place leak testing of charcoal and HEPA filters, the air-aerosol uniform mixing test acceptance criteria were not rigorously met. For this reason, a statistical correction factor will be applied to applicable surveillance test results where required.

3/4.7.7 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, are based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3/4.7.8 HYDRAULIC SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

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3/4.7.5 CONTROL ROOM VENTILATION SYSTEM

CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.1 The control room emergency ventilation system (CREVS) shall be OPERABLE with:

a. Two independent pressurization trains,

b. The charcoal adsorber/HEPA filter unit, and

c. The control room envelope/pressure boundary able to maintain the required positive pressure.

APPLICABILITY: MODES 1, 2, 3, 4, and during the movement of irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3, and 4:

a. With one pressurization train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the filter unit inoperable, restore the filter unit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the control room envelope/pressure boundary inoperable, restore the control room envelope to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

During movement of irradiated fuel assemblies:

d. With one pressurization train inoperable, restore the inoperable pressurization train to OPERABLE status within 7 days, or initiate and maintain operation of the remaining OPERABLE train in the pressurization/cleanup alignment.

e. With any of the following (1) both pressurization trains inoperable; (2) the filter unit inoperable; or (3) the control room envelope/pressure boundary inoperable, immediately suspend all operations involving the movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.5.1 The control room emergency ventilation system shall be demonstrated OPERABLE:

a. Deleted

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the system operates for at least 15 minutes.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

2. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 1.0% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. The carbon samples not obtained from test canisters shall be prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

d. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than ~~or equal to~~ 1.0% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H; or

2. Verifying within 31 days after removal that a laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 1.0% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. and the samples are prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

   a) Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%, and

   b) Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 4 inches Water Gauge (W.G.) while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

2. a. Verifying that on a Safety Injection Signal from Unit 1, the system automatically operates in the pressurization/cleanup mode.

   b. Verifying that on a Safety Injection Signal from Unit 2, the system automatically operates in the pressurization/cleanup mode.

3. Verifying that the system maintains the control room envelope/pressure boundary at a positive pressure of greater than or equal to 1/16 inch W. G. relative to the outside atmosphere at a system flow rate of 6000 cfm plus or minus 10% with a makeup air flow rate of ≤ 1000 cfm.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

CONTROL ROOM AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.2 The Control room air conditioning system (CRACS) shall be OPERABLE with two heating and cooling systems.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one heating and cooling system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.2 The control room air conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is less than or equal to 95°F.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 25,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H; or

2. Verifying within 31 days after removal that laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 5% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. and the samples are prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

SURVEILLANCE REQUIREMENTS (Continued)

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 4 inches Water Gauge while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

2. Deleted.

3. Verifying that the standby fan starts automatically on a Containment Pressure--High-High Signal and directs its exhaust flow through the HEPA filters and charcoal adsorber banks on a Containment Pressure--High-High Signal. +

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

+ The provisions of Technical Specification 4.0.8 are applicable.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. The carbon samples not obtained from test canisters shall be prepared by either:

   (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 30,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that laboratory analyses of at least two carbon samples show a penetration of less than or equal to 5% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. and the samples are prepared by either:

   (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested-in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 4 inches Water Gauge while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

2. Deleted.

3. Verifying that on a high-radiation signal, the system automatically directs its exhaust flow through the charcoal adsorber banks and automatically shuts down the storage pool ventilation system supply fans.

4. Verifying that the exhaust ventilation system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the control room emergency ventilation system (CREVS) ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. In MODES 1-4, the CREVS provides radiological protection to allow operators to take the actions necessary to mitigate the consequences of a design basis accident. The CREVS is also required to be OPERABLE for operations involving the movement of irradiated fuel assemblies to provide protection from a fuel handling accident. The CREVS has two pressurization trains with each pressurization train consisting of a pressurization fan, normal intake air damper, and emergency intake air damper available to align and maintain flow to the control room. The charcoal adsorber/HEPA filter unit consists of the prefilter, charcoal adsorber, HEPA filter, and filter housing. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to less than or equal to 5 rem Total Effective Dose Equivalent, TEDE. This limitation is consistent with the requirements of General Design Criteria (GDC) 19 of Appendix "A", 10 CFR 50.

The control room envelope/pressure boundary consists of the control room, the control room HVAC equipment room, and the plant process computer room and shall be considered OPERABLE if these rooms can be maintained at a positive pressure of greater than or equal to 1/16 inch water gauge relative to the outside atmosphere. If the control room envelope/pressure boundary is inoperable in MODES 1, 2, 3, and 4, actions must be taken to restore an OPERABLE control room boundary within 24 hours. The control room boundary can be opened intermittently under administrative controls. For entry and exit through doors to the control room, the administrative control of opening the pressure boundary is performed by the person entering or exiting the area. For other openings of the control room envelope/pressure boundary, the administrative controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

The Unit 2 control room emergency ventilation system aligns and operates automatically on a Safety Injection (SI) Signal from either Unit 1 or Unit 2. Both pressurization fans start on the SI signal. Procedures direct realignment of the CREVS to single fan operation within two hours after receiving the SI signal. The automatic start from Unit 1 is normally only available when the Unit 1 ESF actuation system is active in modes 1 through 4 in Unit 1.

The Limiting Condition for Operation requires two independent control room heating and cooling systems. Each cooling system requires a functional air handling unit and associated cooling water supply. Cooling water is provided from a chilled water unit. At the design maximum essential service water (ESW) supply temperature of 86°F, a chilled water unit will maintain the control room temperature below 95°F. Cooling water may also be supplied directly by ESW when ESW supply temperature is ≤ 65°F.

The control room air conditioning system (CRACS) normally maintains the control room at temperatures at which control room equipment is qualified for the life of the plant. Continued operation at the Technical Specification limit is permitted since the portion of time the temperature is likely to be elevated is small in comparison to the qualified life of the equipment at the limit.

Each control room cooling system can maintain control room temperature ≤ 102°F during accident conditions with the control room isolated. At control room temperatures of ≤ 102°F, vital control room equipment remains within its manufacturer's recommended operating temperature range.

ATTACHMENT 4 TO C0600-13

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Indiana Michigan Power Company (I&M) has evaluated this proposed amendment and determined that it does not involve a significant hazard. According to 10 CFR 50.92(c), a proposed amendment to an operating license does not involve a significant hazard if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;

2. create the possibility of a new or different kind of accident from any previously evaluated; or

3. involve a significant reduction in a margin of safety.

I&M proposes to amend the Facility Operating Licenses, DPR-58 and DPR-74. I&M proposes to use the methodology and the alternative source term (AST) described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," and draft Regulatory Guide (RG) 1081, "Alternative Radiological Source Terms for Evaluating the Radiological Consequences of Design Basis Accidents at Boiling and Pressurized Water Reactors."

Implementing the AST requires use of a new acceptance criterion for 10 CFR 50, Appendix A, General Design Criterion (GDC) 19, of 5 rem total effective dose equivalent (TEDE). I&M has determined that the revised methodology and acceptance criterion represents an unreviewed safety question. Therefore, NRC review and approval is required prior to implementation.

I&M also proposes to amend Appendix A, "Technical Specifications" (T/S), to DPR-58 and DPR-74. The proposed changes are:

- separating the pressurization/filtration function requirements from the temperature control function in the control room emergency ventilation system (CREVS)
- adding a limiting condition for operation for the control room envelope/pressure boundary and an action statement if the envelope/pressure boundary is inoperable
- modifying the requirements of the CREVS to add a new action, modify a surveillance requirement, and add a definition in the bases for the control room envelope/pressure boundary and an action statement if the envelope/pressure boundary is inoperable
- expanding the applicability requirements and associated actions for CREVS to include, "during the movement of irradiated fuel assemblies"
- modifying action requirements for CREVS to address recently installed redundant dampers

- incorporating the recommendations described in Generic Letter (GL) 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal" into the T/S for the CREVS, engineered safety features ventilation system (ESFVS), and storage pool ventilation system (SPVS)
- reducing the allowable pressure drop across the high-efficiency particulate air (HEPA) filter/charcoal adsorber unit for the CREVS, ESFVS, and SPVS
- making editorial changes to the page format

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The proposed change to implement the AST involves changes to the methodologies and acceptance criterion associated with the control room dose analysis. The actual sequence and progression of accidents are not changed. However, the regulatory assumptions regarding the analytical treatment of the accidents are affected by the change. The use of an AST alone cannot increase the probability of an accident or the core damage frequency. The proposed change to use the AST does not make any changes to equipment, procedures, or processes that increase the likelihood of an accident. It does not affect any accident initiators or precursors. The methodology is used to determine consequences of an accident and has no impact on their likelihood of occurrence. Therefore, this proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The current acceptance criterion specify the dose to personnel in terms of "rem whole body" or equivalent for the duration of the accident, where the dose derived using the AST is given in rem TEDE, as described in 10 CFR 50.67. TEDE includes internal and external exposure; whole body includes external exposure only. The current acceptance criterion focuses on doses to the thyroid and the whole body. It is based on the assumption that the major contributor to dose will be radioiodine. Although this may be appropriate with the Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites", source term implemented by RGs 1.4, it may not be true for a source term based on a more complete understanding of accident sequences and phenomenology. The AST includes a larger number of radionuclides than did the TID-14844 source term as implemented in regulatory guidance. The whole body and thyroid dose criteria considered the noble gases and iodine contributors as the limiting factors. The acceptance criteria of 5 rem TEDE and 5 rem whole body are not equivalent, so they cannot be compared directly. I&M has reanalyzed the loss of coolant accident (LOCA) and non-LOCA events to determine the limiting condition for control room dose using the AST. The calculated dose for all the analyzed events meets the acceptance

criterion for GDC-19 as described in 10 CFR 50.67. Therefore, the consequences are not significantly increased.

The CREVS is designed to mitigate the consequences of an accident. It is not assumed to operate in the pressurization mode until after an accident has occurred. The system itself has no impact on the initiation of any evaluated accidents. Therefore, the changes to the CREVS requirements do not increase the probability of an accident previously evaluated.

The proposed changes to the CREVS requirements do not affect the ability to maintain a control room pressure boundary. The changes ensure that the control room will be pressurized following an accident where the CREVS is required to operate to minimize unfiltered inleakage. The proposed 24-hour allowed outage time and subsequent shutdown action are consistent with the requirements for an inoperable filter unit and are reasonable due to the low probability of the initiation of an accident requiring actuation of the CREVS occurring when the pressure boundary is inoperable. Control room dose is significantly increased with increased unfiltered inleakage. Specifying the test condition in the surveillance allows increases in unfiltered inleakage to be identified and evaluated. Preserving the control room pressure boundary provides assurance that the consequences of an accident previously evaluated are not significantly increased.

The proposed applicability and action requirements during the movement of irradiated fuel assemblies ensure the CREVS is operable for the protection of control room personnel in the event of a fuel handling accident.

The proposed changes to the Limiting Conditions for Operation (LCO) address redundant dampers that are being installed. Adding the new equipment to the LCO and action requirements ensures all components associated with the CREVS are operable or action is taken to restore them. The proposed changes do not affect equipment design or operation. Therefore, the consequences of accidents previously evaluated are not increased.

The proposed changes for the charcoal testing method affect activities in the laboratory only and have no impact on plant operation. Sampling and testing charcoal will not initiate an accident. The charcoal adsorbers are used to mitigate the consequences of an accident and are not operated until after an accident has occurred. Therefore, the probability of an accident previously evaluated is not affected. Charcoal testing verifies the ability of the charcoal adsorbers to function as assumed following an accident. The new method for testing the CREVS samples provides more accurate and reproducible laboratory results. These results provide assurance that the charcoal adsorbers will meet the assumed radioiodine removal efficiency following an accident. Therefore, the consequences of accidents previously evaluated are not increased.

The HEPA filter/charcoal adsorber units in the CREVS, ESFVS, and SPVS are designed to mitigate the consequences of an accident. They are not assumed to operate until after an accident has occurred. The adsorber units have no impact on the initiation of any evaluated accidents.

Therefore, the proposed change to reduce the differential pressure does not increase the probability of an accident previously evaluated. The proposed change to surveillance requirements to reduce the allowable pressure drop across the HEPA filter/charcoal adsorber unit ensures the system flow rates can be maintained so that the system performs as designed. The change ensures that filter units are replaced before airflow is restricted. This allows the required area to be pressurized so that unfiltered inleakage remains within the amount assumed in the accident analysis. Therefore, the proposed revision to reduce the allowable pressure drop requirement does not significantly increase the consequences of an accident previously evaluated.

The remaining changes are administrative in nature. The proposed editorial changes involve reformatting of the individual T/S pages to standardize page appearance and readability and do not alter any requirements. The proposed change to separate the CREVS functions into individual specifications does not affect the system operability requirements or make any changes in how the equipment is operated. The separation of the two functions does not affect the ability of the CREVS to cool or pressurize the control room envelope. The proposed change to incorporate the new laboratory testing standard for charcoal adsorbers in the ESFVS and SPVS is administrative because the test conditions are consistent with the standard referenced in the T/S. These changes are administrative in nature and do not affect the probability or consequences of accidents previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The use of an AST alone cannot create the possibility of a new or different kind of accident. The proposed change to use the AST does not make any changes to equipment, procedures, or processes. The AST does not create any new accident initiators or precursors. It is merely a method used to predict radionuclides released following an accident. Therefore, this proposed change does not increase the possibility of a new or different kind of accident than previously evaluated.

The CREVS is designed to mitigate the consequences of an accident. It is not assumed to operate until after an accident has occurred. The proposed LCO requirement to maintain the control room envelope/pressure boundary operable and expand the area to include the control room heating ventilation and air conditioning equipment room and plant process computer room does not affect system design or operation. The area defined as the control room envelope includes all of the areas that communicate with the control room. A tracer gas test confirmed that the defined control room envelope can be pressurized to greater than or equal to 1/16 inch of water gauge, as assumed in the accident analysis. The proposed surveillance requirement to specify a makeup airflow rate of less than or equal to 1000 cubic feet per minute (cfm) allows periodic verification that the assumed unfiltered inleakage is within the assumptions of the accident analysis. The new requirement provides added assurance that the pressure boundary is

maintained operable. The proposed changes to the CREVS requirements do not introduce any new plant equipment or new methods of operating the equipment. No new failure mechanisms are introduced.

The proposed change to incorporate the new testing requirements of ASTM D3803-1989 is administrative in nature. It affects activities in the laboratory only and has no impact on plant operation. The change does not affect the method for obtaining the charcoal sample. It does not cause any of the ventilation equipment to be operated in a new or different manner.

The change to reduce the allowable pressure drop across the pressurization filter train to 4 inches water gauge ensures system performance is consistent with design. The revised value is more restrictive and provides assurance that the affected components of the filter unit are replaced before airflow is reduced to the extent that it affects the pressurization capability of the CREVS, ESFVS, and SPVS. No new failure mechanism is created.

The remaining changes are administrative in nature. The proposed editorial changes involve reformatting of the individual T/S pages to standardize page appearance and readability and do not alter any requirements. The proposed change to separate the CREVS functions into individual specifications does not affect the system operability requirements or make any changes in how the equipment is operated. The separation of the two functions does not affect the ability of the CREVS to cool or pressurize the control room envelope. The proposed change to incorporate the new laboratory testing standard for charcoal adsorbers in the ESFVS and SPVS is administrative because the test conditions are consistent with the standard referenced in the T/S. These changes are administrative in nature and do not create the possibility of a new or different kind of accident from any previously evaluated.

Therefore, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change to implement the AST for the revised analysis incorporates the guidance for application of the AST provided in NUREG-1465 and draft RG-1081. The change involves the use of new terminology for the acceptance criterion expressed as 5 rem TEDE. The term TEDE is defined in 10 CFR 20 as the sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). The acceptance criteria of 5 rem TEDE and 5 rem whole body are not equivalent. The NRC has revised the current GDC-19 whole body dose criterion with a criterion in terms of rem TEDE for the duration of the accident in 10 CFR 50.67 for the licensee that seeks to revise its current radiological source term with an AST.

The NRC recognizes that an analysis using the AST may represent a reduction in the margin of safety for some applications. The margin of safety is typically defined as the difference between the calculated parameters (offsite and control room dose) and the associated regulatory or safety limit. Implementing the AST in accordance with draft RG-1081 and 10 CFR 50.67 revises the acceptance criterion (regulatory limit) contained in GDC-19 to 5 rem TEDE. The calculated control room dose is below the new acceptance criterion. In 10 CFR 50.67, the rule considers the 5 rem whole body, or its equivalent to any part of the body is accounted for in the definition of TEDE and by the 5 rem TEDE annual limit. Therefore, revising the control room dose analysis using the new terminology for the AST does not involve a significant reduction in a margin of safety.

The margin of safety associated with the CREVS T/S is to maintain control room dose within the limits of GDC-19. The proposed changes to the CREVS requirements ensure that accident analysis assumptions are preserved so that the dose limit is met. The proposed change for control room envelope/pressure boundary provides assurance that positive pressure is maintained in the envelope and that unfiltered inleakage is bounded by the accident assumption. Adding a test requirement for filtered makeup airflow also supports this requirement. The proposed change to expand the applicability requirements and actions provides assurance that the CREVS is operable during times when an accident could occur that may affect the control room environment. The proposed changes that reflect addition of the dampers provides assurance that the control room pressure boundary will be isolated and the envelope will be pressurized when CREVS is actuated following an accident. The proposed change to reduce the allowable pressure drop across the HEPA filter/charcoal adsorber units provides assurance that the CREVS, ESFVS, and SPVS provide the required airflow. This allows areas to be pressurized as required.

The proposed change to incorporate the testing standards recommended for the charcoal adsorbers in GL 99-02 provides assurance that the charcoal adsorbers will remove radioiodine as assumed in the accident analysis. Additional margin is gained by applying a safety factor to the iodine removal efficiency assumed in the accident analysis. This safety factor applies to CREVS, ESFVS, and SPVS. The T/S have also been revised to reflect the iodine removal efficiency assumed in the accident analysis. The acceptance criterion reflects the analysis assumption and the safety factor.

The remaining changes are administrative in nature. The proposed editorial changes involve reformatting of the individual T/S pages to standardize page appearance and readability and do not alter any requirements. The proposed change to separate the CREVS functions into individual specifications does not affect the system operability requirements or make any changes in how the equipment is operated. The separation of the two functions does not affect the ability of the CREVS to cool or pressurize the control room envelope. The proposed change to incorporate the new laboratory testing standard for charcoal adsorbers in the ESFVS and SPVS is administrative because the test conditions are consistent with the standard referenced in

the T/S. These changes are administrative in nature and do not involve a significant reduction in the margin of safety.

The proposed changes support the control room dose calculations that demonstrate that the GDC-19 requirement will be met. Therefore, these changes do not involve a significant reduction in the margin of safety.

In summary, based upon the above evaluation, I&M has concluded that these changes involve no significant hazards consideration.

ATTACHMENT 5 TO C0600-13

ENVIRONMENTAL ASSESSMENT

Indiana Michigan Power Company (I&M) has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. I&M has determined that this license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Attachment 4, this proposed amendment does not involve significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed amendment involves a revision to control room dose analysis and changes to the Technical Specifications (T/S) affecting requirements for the control room emergency ventilation system, auxiliary building engineered safety features ventilation system (ESFVS), and spent fuel storage pool ventilation system (SPVS). The auxiliary building ESFVS and spent fuel (SPVS) directly interface with radioactive and nonradioactive effluent processing and control systems. However, these proposed changes do not result in the generation of any additional radioactive or nonradioactive effluents. In addition, the proposed changes to the T/S increase the effectiveness of these radioactive and nonradioactive effluent processing and control systems. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes do not result in significant changes in the operation or configuration of the facility. There is an increase in the effective level of controls and methodology used for processing of radioactive effluents, and the proposal does not result in any change in the normal radiation levels within the plant. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure resulting from this change.

ATTACHMENT 6 TO C0600-13

Final Licensing Report for Control Room Radiological Consequences of Accidents for the Donald C. Cook Nuclear Plant Units 1 and 2 Using Source Term Methodology from NUREG-1465

Westinghouse
Electric Company LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

AEP-00-072
February 28, 2000

Mr. Jeb Kingseed
American Electric Power
500 Circle Drive
Buchanan, Michigan 49107

AMERICAN ELECTRIC POWER
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
Licensing Report for the Radiological Consequences of Accidents Using NUREG-1465 Source term Methodology

Dear Mr. Kingseed,

Attachment A to this letter is the Licensing Report for the NUREG-1465 Source Term Methodology Radiological Consequences of Accidents for the Donald C. Cook Nuclear Plant Units 1 and 2. This report supercedes the "revised final" licensing report issued via AEP-99-477. The analyses described in the report, use or bound the assumptions provided in Reference 1 of Attachment A. The site boundary and low population zone atmospheric dispersion factors provided in Reference 1 of Attachment A are listed as "unverified". Attachment B to this letter contains a copy of Reference 1.

This work was performed under AEP contract C-7693 Release 00-02 (SATR-00-003). Please contact Mr. Don Peck (412-374-2052) or me if you have further questions on this subject.

Donald E. Peck /bar
W. R. Rice
Customer Projects Manager

DEP
Attachment

cc: Rick Kohrt - AEP

LTR-CRA-00-47

From : Containment and Radiological Analysis
WIN : 284-4454
Date : February 28, 2000
Subject : Licensing Report for the Radiological Consequences of Accidents Using NUREG-1465 Source Term Methodology for D. C. Cook Units 1 and 2

To: D. E. Peck

cc: E. C. Arnold*
J. L. Grover*
A. E. Durham*
* without attachment

Attachment 1 to this letter is the licensing report for the NUREG-1465 Source Term Methodology Radiological Consequences of Accidents for D. C. Cook Units 1 and 2. This report supercedes the "revised final" licensing report issued in AEP-99-477 (SAE-CRA-99-348). The analyses described in the report, use or bound the assumptions provided in Reference 1. The radiological analyses which support the attached licensing report are documented in References 2 to 15. The site boundary and low population zone atmospheric dispersion factors provided in Reference 1 are listed in Reference 1 as "unverified". These "unverified" assumptions have been used in the calculation of the Reference 2 through 15 dose analyses.

If you have any questions, please contact the undersigned.

U. Bachrach
Containment and
Radiological Analysis

Jill S. Monahan

J. S. Monahan*
Containment and
Radiological Analysis

Official record electronically approved in EDMS 2000

List of References

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2. Westinghouse Calculation Note, CN-CRA-99-047, Revision 0, "D. C. Cook Units 1 & 2 Steam Releases for Radiological Dose Calculation," 7/99.
3. Westinghouse Calculation Note, CN-REA-99-30, Revision 0, "AEP/AMP Accident Source Terms," 7/99.
4. Westinghouse Calculation Note, CN-CRA-99-54, Revision 0, "D. C. Cook - Definition of Iodine Spike Rate and Duration," 7/99.
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9. Westinghouse Calculation Note, CN-CRA-99-55, Revision 1, "Donald C. Cook Steam Generator Tube Rupture T&H Analysis for NUREG-1465 Dose Project - Revised," 9/99.
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12. Westinghouse Calculation Note, CN-CRA-99-66, Revision 0, "D. C. Cook - Fuel Handling Accident Doses," 9/99.
13. Westinghouse Calculation Note, CN-CRA-99-68, Revision 1, "D. C. Cook - Main Steam Line Break Doses," 11/99.
14. Westinghouse Calculation Note, CN-CRA-99-60, Revision 2, "D. C. Cook (AEP/AMP) Loss of Load Radiation Dose Analysis," 11/99.
15. Westinghouse Calculation Note, CN-CRA-99-107, Revision 0, "D. C. Cook Gas Decay Tank Rupture & Volume Control Tank (VCT) Rupture Radiation Dose Analysis," 11/99.

Attachment 1 to LTR-CRA-00-47

Licensing Report
for the
Radiological Consequences of Accidents
Using NUREG-1465 Source Term Methodology
for D. C. Cook Units 1 and 2

Final Licensing Report for Control Room Radiological Consequences of Accidents for the Donald C. Cook Nuclear Plant Units 1 and 2 Using Source Term Methodology from NUREG-1465

Prepared for American Electric Power
by Westinghouse Electric Company, LLC
February 28, 2000

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1.0 RADIOLOGICAL CONSEQUENCES UTILIZING NUREG-1465 SOURCE TERMS

1.1 Introduction

The Donald C. Cook licensing basis for the radiological consequences analyses for Chapter 14 of the UFSAR is currently based on methodologies and assumptions that are derived from TID-14844 (Reference 8) and other early guidance.

NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 1) provides a postulated fission product source term that is based on the current understanding of light-water reactor (LWR) accidents and fission product behavior. Reference 1 is applicable to LWR designs and is intended to form the basis for the development of regulatory guidance 1081 (Reference 11).

The new source terms as described in NUREG-1465 (Reference 1) are being used to calculate the control room radiological consequences for the Donald C. Cook Nuclear Plants to support the control room habitability program. To support the control room habitability, the following UFSAR Chapter 14 radiological consequences analyses will be analyzed or evaluated: Large Break Loss-of-Coolant Accident (LBLOCA), Small Break Loss-of-Coolant Accident (SBLOCA), Steam Generator Tube Rupture (SGTR), Locked Rotor, Rod Ejection, Fuel Handling Accident (FHA), Main Steamline Break (MSLB), Waste Gas Tank Ruptures, and Loss of Offsite Power. Each accident and the specific input assumptions are described in detail in subsequent sections in this report.

1.2 Common Analysis Inputs and Assumptions

The assumptions and inputs described in this section are common to analyses discussed in this report. The accident specific inputs and assumptions are discussed in Sections 2 through 10.

The total effective dose equivalent (TEDE) doses are determined for control room personnel (CR). The dose conversion factors (DCFs) used in determining the committed effective dose equivalent (CEDE) or inhalation dose are from Reference 2. The TEDE dose is equivalent to the CEDE dose plus the acute dose for the duration of exposure to the cloud. The CEDE dose DCFs are given in Table 1.

The γ-body (acute) doses are based on the average disintegration energies from Reference 2 for the iodine isotopes and from Reference 5 for the remainder of the nuclides (except the noble gases). The dose conversion factors for the noble gases are taken from ICRP Publication 30 (Reference 4). The average disintegration energies and dose conversion factors for the γ-body doses are listed in Table 2.

Parameters modeled in the control room personnel dose calculations are provided in Table 3. These parameters include the normal operational flowrates, the emergency operation flowrates, control room volume, filter efficiencies and control room operator breathing rates. For each event the limiting control room volume and time to turn off the second pressurization fan were used, as confirmed by sensitivity studies. The control room atmospheric dispersion factors are provided for each accident individually since the different events have separate release points relative to the control room HVAC intake. The control room personnel dose includes the direct dose from the radiation cloud outside the control room as well as the inhalation and acute doses from the activity introduced inside the control room. The direct dose takes into account the shielding afforded by the control walls which are 18 inch thick concrete.

The current control room personnel dose calculations use the iodine protection factor methodology from Murphy-Campe (Reference 19) which expresses the concentration of radioiodines in the control room as a fixed fraction of the concentration at the control room outdoor air intakes. The calculations for the analyses presented in this report do not use this model. Instead the control room is modeled as a discrete volume. The atmospheric dispersion factors calculated for release of activity from the event specific release point to the control room intake are used to determine the activity available at the intake. The inflow (filtered and unfiltered) to the control room and the control room recirculation flow listed in Table 3 are used to calculate the activity introduced to the control room and cleanup of activity from that flow.

The core fission product activity is provided in Table 4 for all nuclides. The Technical Specification nominal reactor coolant activity based on 1% fuel defects for noble gases and 1.0 μCi/gm Dose Equivalent I-131 is provided in Table 5. Decay constants for each nuclide are provided in Table 6.

2.0 LARGE BREAK LOSS OF COOLANT ACCIDENT

An abrupt failure of the main reactor coolant pipe is assumed to occur and it is assumed that the emergency core cooling features fail to prevent the core from experiencing significant degradation (i.e., melting). This sequence cannot occur unless there are multiple failures and thus goes beyond the typical design basis accident that considers a single active failure. Activity from the core is released to the containment and from there released to the environment by means of containment leakage, leakage from the emergency core cooling system or purging of the containment atmosphere.

2.1 Comparison of NUREG-1465 Source Term Methodology to TID-14844

The reanalysis of the LBLOCA control room doses for D. C. Cook Units 1 and 2 uses the following NUREG-1465 source term characteristics in place of those identified in TID-14844:

Iodine chemical species
Fission product release timing
Fission product release phases through early in-vessel
Fission product release fractions
Fission product groups

A comparison of NUREG-1465 to TID-14844 is provided in Tables 7 through 10.

2.2 Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 11. Activity from the reactor coolant system and melted fuel is released into the containment. The analysis considers the release of activity from the containment via containment leakage and leakage from external recirculation. In addition releases are modeled from the containment purge system assumed to be operating at the start of the event and releases due to a passive failure in the ECCS system assumed to occur at 24 hours into the event. The following sections address topics of significant interest.

The total control room dose is the sum of the doses resulting from each of the postulated release paths and nuclides considered.

2.2.1 Source Term

The reactor coolant activity is assumed to be released over the first 30 seconds of the accident. However, the activity in the coolant is insignificant compared with the release from the core and is not included in the analysis. The exception to this is the purge release case where only the activity in the coolant is assumed to be available for release.

The use of NUREG-1465 (Reference 1) source term modeling results in several major departures from the assumptions used in the existing LOCA dose analysis as reported in the FSAR. Instead of assuming instantaneous melting of the core and release of activity to the containment, the release of activity from the gap and melted core occurs over a 1.8 hour interval.

Instead of considering only the release of iodines and noble gases, a wide spectrum of nuclides is taken into consideration. Table 10 lists the nuclides being considered for the LOCA with core melt (eight groups of nuclides). Tables 8 and 9 provide a comparison between the fission product release fractions and the timing/duration of releases to the containment as assumed in TID-14844 (Reference 8) and in NUREG-1465 (Reference 1).

Instead of the iodine being primarily in the elemental form, the iodine is mainly in the form of cesium iodide which exists as particulate and the fraction that is in the organic form is much smaller. The iodine characterization from NUREG-1465 is compared in Table 7 with that from Regulatory Guide 1.4.

The other groups of nuclides (other than the iodines and the noble gases) all occur as particulates only.

For the containment leakage analysis all activity released from the fuel is assumed to be in the containment atmosphere until removed by sprays, ice condensers, radioactive decay or leakage from the containment. For the ECCS leakage and passive failure analyses all iodine activity released from the fuel is assumed to be in the sump solution until removed by radioactive decay or leakage from the ECCS. For the containment purge analysis the iodine and noble gases present in the reactor coolant system prior to the LOCA are assumed to be in the containment atmosphere and released via the containment purge system until it is isolated.

2.2.2 Containment Modeling

The containment is modeled as 3 discrete volumes: upper containment, active lower containment and annulus region (also referred to as the lower containment dead end region). The volumes are conservatively assumed to be mixed by the flowrate through the ice condenser bed and a conservative flowrate to the annulus region. The containment leak rate is not dependent on the volume size.

The containment is assumed to leak at the design leak rate of 0.25 wt% per day for the first 280 hours of the accident and to leak at half that rate (0.125 wt% per day) after 280 hours. The reduction in leak rate at 280 hours is inconsistent with Regulatory Guide 1.4 which specifies a long term leak rate equal to one half the initial rate after 24 hours. The containment leakage analyses do not model the time required for isolation of the containment. Since there would be very little activity in the containment that early in the event. The release of activity from the early part of the event where the reactor coolant activity has been released to containment is conservatively modeled by assuming that the purge system is operating, providing a maximum flow from the containment (See Section 2.2.5).

The flow between the active lower containment and the upper containment is provided by the deck fans. Operation of these fans is conservatively delayed until 3 minutes into the event.

The flow between the active lower containment and the dead end region is unknown. For the analysis it was set to 500 cfm. Sensitivity analyses showed that this flow rate was more limiting than 100 cfm or 900 cfm. As the flow is reduced less activity is transferred into the dead end region, where it is not subject to removal by the sprays. (Spray removal is conservatively neglected in the dead end region in the model used for the analysis). As the flow increases the activity tends to be flushed out of the dead end volume back into the region where it is subject to removal by the sprays. The flow rate chosen, together with the assumption that there is no removal in the dead end region due to sprays, plateout or sedimentation, provides a conservative analysis.

2.2.3 Removal of Activity from the Containment Atmosphere

The removal of elemental iodine and particulates from the containment atmosphere is accomplished only by containment sprays, ice condensers and radioactive decay. The noble gases and the organic iodine are subject to removal only by radioactive decay.

One train of the containment spray system is assumed to operate following the LOCA. When the RWST drains to a predetermined setpoint level the operators switch to recirculation of sump liquid to provide a source for the sprays. The switchover is assumed to take 5 minutes. During this 5 minutes the analysis does not credit any spray removal in the containment. Sensitivity analyses determined that (for the NUREG-1465 release model) a delayed time to switchover to recirculation is conservative. The analysis conservatively assumed that the switchover is initiated at 1.25 hours from the start of the event. This time bounds the longest time to switchover calculated based on RWST drain down calculations performed assuming a single train of ECCS in operation and minimum suction flow rates. This time for switchover delays the 5 minutes with no spray removal credited until the early in-vessel release phase (i.e., a point in the event when the containment activity is highest).

2.2.3.1 Containment Spray Removal of Elemental Iodine

The current Standard Review Plan (Reference 10) identifies a methodology for the determination of spray removal of elemental iodine independent of the use of spray additive. The removal rate constant is determined by:

$$\lambda_s = 6K_gTF / VD$$

where λ_s = Removal rate constant due to spray removal, hr^{-1}
K_g = Gas phase mass transfer coefficient, ft/min
T = Time of fall of the spray drops, min
F = Volume flow rate of sprays, ft^3/hr
V = Containment sprayed volume, ft^3
D = Mass-mean diameter of the spray drops, ft

The upper limit specified for this model is 20 hr^{-1}.

Parameters for D. C. Cook are listed below and were chosen to bound the current plant configuration:

Upper Containment	Active Lower Containment
K_g = 9.84 ft/min	Kg = 9.84 ft/min
T = 4.9 sec	T = 2.0 sec
F = 1950 gpm	F = 700 gpm
V = 9.9×10^5 ft^3	V = 3.3×10^5 ft^3
D = 0.0999 cm	D = 0.103 cm

These parameters and the appropriate conversion factors were used to calculate the elemental spray removal coefficients. Conservative elemental removal coefficients used in the analysis are listed in Table 11. The values in Table 11 have been reduced by an additional 10% for conservatism.

The annulus region has a very small spray fall height of approximately 5 ft. The analysis conservatively does not take credit for removal of elemental iodine in the annular region.

When sprays are operating in the recirculation phase the elemental removal coefficients are reduced to address the loading of the recirculating solution with elemental iodine. Conservative elemental removal coefficients used in the analysis are listed in Table 11.

Removal of elemental iodine from the containment atmosphere is assumed to be terminated when the airborne inventory drops to 0.5 percent of the total elemental iodine released to the containment (this is a DF of 200). With the NUREG-1465 source term methodology this is interpreted as being 0.5% of the total inventory of elemental iodine that is released to the containment atmosphere over the duration of gap and in-vessel release phases. In the analysis this occurs after 2.78 hours. After that time removal of elemental iodine from the containment atmosphere is no longer modeled.

The analysis does not take credit for elemental iodine removal by the RHR sprays in the upper containment.

2.2.3.2 Containment Spray Removal of Particulates

Particulate spray removal is determined using the model described in Reference 10.

The first order spray removal rate constant for particulates may be written as follows:

$$\lambda p = 3hFE / 2VD$$

where h = Drop Fall Height
F = Spray Flow Rate
V = Volume Sprayed
E = Single Drop Collection Efficiency
D = Average Spray Drop Diameter

Parameters for D. C. Cook are listed below and were chosen to bound the current plant configuration:

Upper Containment
h = 59 ft
F = 1950 gpm
V = 9.9×10^5 ft^3

Active Lower Containment
h = 26 ft
F = 700 gpm
V = 3.3×10^5 ft^3

RHR Spray
h = 41 ft
F = 1850 gpm
V = 9.9×10^5 ft^3

The E/D term depends upon the particle size distribution and spray drop size. From Reference 10 it is conservative to use 10 m^{-1} for E/D until the point is reached when the inventory in the atmosphere is reduced to 2% of its original (DF of 50). With the NUREG-1465 source term methodology this is interpreted as being 2% of the total inventory particulate iodine that is released to the containment atmosphere over the duration of gap and in-vessel release phases.

These parameters and the appropriate conversion factors were used to calculate the particulate spray removal coefficients. Conservative particulate removal coefficients used in the analysis are listed in Table 11. The values in Table 11 have been reduced by an additional 10% for conservatism. RHR spray to the upper containment is credited starting after 1.25 hours plus 5 minutes or 1.333 hours. When the airborne inventory drops to 2 percent of the total particulate

iodine released to the containment (this is a DF of 50) these removal coefficients are reduced by a factor of 10. In the analysis this occurs just before 2.41 hours.

The analysis assumes that recirculation spray and RHR spray continue until 6 hours from the start of the event.

The annulus region has a very small spray fall height of approximately 5 ft. The analysis conservatively does not take credit for removal of particulate iodine in the annular region.

2.2.3.3 Containment Ice Condenser Removal of Elemental Iodine

Elemental iodine removal by the ice condensers is also credited, with a removal efficiency of 0.3, starting when the deck fans start (Reference 18). This is modeled as a filter on the deck fan flow from the active lower containment to the upper containment. Ice condenser iodine removal is credited until a DF of 200 in the containment inventory of elemental iodine is reached, or until the first ice bed melts through (Reference 18), whichever is shorter. The analysis conservatively limits ice removal of elemental iodine to the first hour of the event. The DF of 200 is not reached until after 2 hours as provided previously. The ice condenser removal of 0.3 efficiency is not significant in this calculation since it represents a small removal in comparison to the spray removal of the elemental iodine.

2.2.4 ECCS Leakage

Activity contained in the sump is recirculated, and may leak outside containment. The analysis considers a leak of 0.2 gpm outside containment. Only 0.01% of the iodine contained in this leakage is assumed to become airborne. No filtration is credited on the release of this airborne activity from the point of leakage at the Unit Vent to the Control Room Intake.

The concentration of activity in the recirculating sump solution is determined based on the calculated sump volume over the course of the transient, with a conservatively slow contribution of melted ice to the sump. Although recirculation is not initiated until the RWST has drained to the pre-determined setpoint level the analysis conservatively considers this leakage from the start of the event.

For the analysis all iodine activity is assumed to be elemental in form. In the NUREG-1465 source term methodology the iodine is mainly in the form of cesium iodide which exists as particulate and would not become airborne. However, to bound a condition where the recirculation liquid was in a low pH environment and changed form, all of the leakage is assumed to be elemental.

2.2.5 Containment Purge

The containment purge system may be in operation at the time when the LOCA occurs, with a flow rate of 16000 cfm. The purge system is isolated within 5 seconds of event initiation. The activity available for release during this time is limited to the iodine and noble gas activity initially in the reactor coolant system. The analysis conservatively delayed purge isolation until 10 seconds.

2.2.6 Passive Failure

D. C. Cook does not have an ESF filter on all portions of the auxiliary building. Therefore, a passive failure in the ECCS recirculation system resulting in a leakage of 50 gpm for half an hour starting 24 hours in to the event is considered. With the exception of the leakage assumption this is identical to the ECCS leakage case discussed above.

2.3 Acceptance Criteria

The criterion for the control room dose are 5 rem TEDE per Draft 10CFR50.67 (Reference 12).

2.4 Results and Conclusions

The large break LOCA control room dose is 2.0 rem TEDE.

The acceptance criteria is met.

3.0 SMALL BREAK LOSS OF COOLANT ACCIDENT

This event is not part of the existing licensing basis for D. C. Cook. This event is considered to be bounded by the large break LOCA event since spray actuation will occur for small breaks for typical ice condenser plants. However D. C. Cook has indicated that there is a possibility that the operators will terminate containment spray for small break LOCA events. Therefore, a small break LOCA analysis has been analyzed here without any credit for iodine or particulate removal by the containment spray system.

An abrupt failure of the primary coolant system is assumed to occur and it is assumed that the break is small enough that the containment spray system is not actuated by high containment pressure but that the core experiences some cladding damage such that the fission product gap activity of damaged fuel rods is released. The analysis conservatively assumes that the gap activity of all rods is released. Activity that is released to the containment is assumed to be released to the environment due to the containment leaking at its design rate. There is also a release path through the steam generators (primary to secondary leakage) until the primary system becomes depressurized to below the secondary system pressure.

3.1 Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 12. The following sections address topics of significant interest that benefit from extended discussion.

3.1.1 Source Term

The use of NUREG-1465 (Reference 1) source term modeling results in a gap fraction of 5.0% being used (consistent with the large break LOCA). This is an appropriate assumption when all fuel rods are assumed to fail. Only the iodines, noble gases, and alkali metals (cesium & rubidium) are assumed to be available for release from the gap. The gap release is consistent with the release timing in Table 8. The coolant activity at 60 μCi/gm is initially in the RCS. 3% of the gap fraction is released from 30 seconds to 90 seconds and the remaining 2% of the gap fraction is released over the next 28.5 minutes. This is consistent with the gap release in the first 30 minutes of the event (Table 8).

The contribution from the primary and secondary coolant inventories prior to the accident is included even though it is trivial in comparison to the gap release from the entire core.

For the primary to secondary leakage pathway, the activity released from the fuel is conservatively assumed to remain in the primary coolant (transfers to the containment are ignored) and available to leak into the secondary coolant.

3.1.2 Iodine Chemical Form

NUREG-1465 specifies that the iodine released from the fuel is in the form of 95% cesium iodide, 4.85% elemental, and 0.15% organic. These fractions are used for the containment leakage release pathway. For the primary to secondary leakage pathway, the guidance of draft regulatory guide DG-1081 (Reference 11) states that the iodine in this case is assumed to be all in the elemental form but, after being released to the atmosphere it is assumed to be 97% elemental and 3% organic. Since the Donald C. Cook control room filtration system has the same efficiency for elemental and organic iodine, the iodine is modeled as all elemental in the primary to secondary leakage case for simplification.

3.1.3 Release Pathways

Conservatively, all the iodine, alkali metals group and noble gas activity (from prior to the accident and resulting from the accident) is assumed to be in the primary coolant (and not in the containment) when determining doses due to the primary to secondary steam generator tube leakage. The primary to secondary tube leakage and steaming from the steam generators continues until the reactor coolant system pressure drops below the secondary pressure. A bounding time of 2 hours was selected for this analysis, although the current analyses of record for the small break LOCA peak cladding temperature have shown in the pressure transient curves that this would occur well before that time.

The primary to secondary steam generator tube leak used in the analysis is a total of 1 gpm consistent with the Technical Specification for Unit 2. Although the primary to secondary pressure differential drops throughout the event, the constant flow rate is maintained.

When determining the doses due to containment leakage, all of the iodine, alkali metal and noble gas activity is assumed to be in the containment. The design basis containment leak rate of 0.25 wt% per day is used for the initial 280 hours. Thereafter, the containment leak rate is assumed to be one-half the design value, or 0.125 wt% per day (Reference 7).

3.1.4 Removal Coefficients

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

An iodine partition factor in the steam generators of 0.01 curies/gm steam per curies/gm water is used (Reference 15). This partition factor is also used for the alkali metal activity in the steam generators. This is highly conservative for the alkali metal activity since the particulates will not readily be removed by the steam phase.

For the containment leakage pathway, credit is taken for sedimentation removal of particulates. Based on the Containment Systems Experiments (CSE) which examined the air cleanup experienced through natural transport processes, it was found that a large fraction of the aerosols were deposited on the floor rather than on the walls indicating that sedimentation was the dominant removal process for the test (Reference 17). The CSE tests determined that there was a significant sedimentation removal rate even with a relatively low aerosol concentration. From Reference 17, even at an air concentration of 10 $\mu g/m^3$, the sedimentation removal coefficient was above 0.3 hr^{-1}. The total concentration of the cesium released to the containment atmosphere, would be in excess of 150,000 $\mu g/m^3$ and an even higher sedimentation rate would be expected. The sedimentation removal coefficient is conservatively assumed to be only 0.1 hr^{-1}. It is also conservatively assumed that sedimentation removal does not continue beyond a DF of 1000. Sedimentation is credited for 65 hours, at which time the DF of 1000 has not been reached.

It is assumed that the containment spray system is not actuated (operation of the containment spray system would more rapidly remove airborne particulates and elemental iodine).

No credit is taken for any removal of elemental iodine in containment via sprays, ice condensers or sedimentation.

3.2 Acceptance Criteria

The criterion for the control room dose are 5 rem TEDE per Draft 10CFR50.67 (Reference 12).

3.3 Results and Conclusions

The small break LOCA control room dose is 1.6 rem TEDE

The acceptance criteria is met.

4.0 STEAM GENERATOR TUBE RUPTURE

The steam generator tube rupture (SGTR) event is separated into two analyses, a thermal and hydraulic analysis and a radiological consequences analysis. Both are discussed in this section.

4.1 Steam Generator Tube Rupture Thermal and Hydraulic Analysis

4.1.1 Introduction

In support of the D. C. Cook NUREG-1465 doses, a steam generator tube rupture (SGTR) thermal-hydraulic analysis for calculation of the radiological consequences has been performed. The SGTR analysis supports a T_{avg} window range of 547°F up to 576.3°F for Unit 1 and a T_{avg} window range of 547°F up to 581.3°F for Unit 2. Plant secondary side conditions (e.g., steam pressure, flow, temperature) are based on high and low tube plugging (0% up to 30% for Unit 1 and up to 15% peak for Unit 2) to bound all possible conditions. Therefore, many separate cases have been analyzed to cover all conditions.

The major hazard associated with an SGTR event is the radiological consequences resulting from the transfer of radioactive reactor coolant to the secondary side of the ruptured steam generator (SG) and subsequent release of radioactivity to the atmosphere. Acceptance criteria for radiological consequences are expressed as maximum allowed Total Effective Dose Equivalent doses as defined in Draft Rule 10CFR50.67 (Reference 12). The primary thermal-hydraulic parameters which affect the calculation of doses for an SGTR include the amount of reactor coolant transferred to the secondary side of the ruptured steam generator, the amount of primary to secondary break flow that flashes to steam and the amount of steam released from the ruptured steam generator to the atmosphere.

The accident analyzed is the double-ended rupture of a single SG tube. It is assumed that the primary-to-secondary break flow following an SGTR results in depressurization of the reactor coolant system (RCS), and that reactor trip and safety injection (SI) are automatically initiated on low pressurizer pressure. Loss of offsite power (LOOP) is assumed to occur at reactor trip resulting in the release of steam to the atmosphere via the steam generator atmospheric relief valves (ARVs) and/or safety valves. Following SI actuation, it is assumed that the RCS pressure stabilizes at the value where the SI and break flowrates are equal. The equilibrium primary-to-secondary break flow is assumed to persist until 30 minutes after the initiation of the SGTR, at which time it is assumed in the analysis that the operators have completed the actions necessary to terminate the break flow and the steam releases from the ruptured steam generator.

After 30 minutes, it is assumed in the analysis that steam is released only from the intact steam generator in order to dissipate the core decay heat and to subsequently cool the plant down to the residual heat removal (RHR) system operating conditions. It is assumed that plant cooldown to RHR operating conditions is not accomplished until 30 days after initiation of the SGTR and that steam releases are terminated at that time. This should bound any extended cooldown to establish RHR. A primary and secondary side mass and energy balance is used to calculate the steam release and feedwater flow for the intact steam generator from 0 to 2 hours, from 2 to 8 hours, from 8 to 24 hours, from 1 to 7 days and from 7 to 30 days.

4.1.2 Input Parameters and Assumptions

The SGTR input data for the NUREG-1465 program is documented in Reference 7. A summary of key input assumptions for the SGTR event follows.

Safety Injection Flowrates

A larger SI flowrate results in a greater RCS equilibrium pressure and, consequently, higher break flow. Maximum flowrates were therefore assumed for this analysis.

RHR Cut-in Time

The RHR cut-in time is modeled in the SGTR analysis. This cut-in time affects the duration of long-term steam releases from the intact steam generator to the atmosphere following termination of the break flow. The effect of RHR cut-in time on long-term doses, however, is not significant since the radiation released from the intact steam generator is small relative to that released by the ruptured steam generator. A bounding RHR cut-in time of 30 days has been assumed. The RHR cut-in pressure and temperature has been conservatively chosen as 212°F and 14.7 psia.

Miscellaneous Parameter Assumptions

Low pressurizer pressure SI actuation setpoint = 1830 psia for Unit 1 and 1915 psia for Unit 2. Lowest SG safety valve reseat pressure = 888 psia, includes 18% Main Steam Safety Valve (MSSV) blowdown which covers the -3% safety valve setpoint tolerance.

4.1.3 Description of Analyses Performed

Break Flow, Steam Releases and Feedwater Flows

Cases were considered in the SGTR thermal-hydraulic analysis to bound the operating conditions for the two D. C. Cook Units for RCS temperature, pressure and steam generator tube plugging ranges. Note that these cases are individually analyzed in order to determine the limiting steam release and limiting break flow between 0 and 30 minutes (break flow termination) for the radiological consequences calculation. A single calculation is performed to determine long-term steam releases from, and feedwater flow to, the intact steam generator for the time interval from the start of the event (0 hours) to 2 hours and from 2 hours to RHR cut-in at 30 days. The 0 to 2 hour calculations use the 0 to 30 minute intact steam generator steam release and feedwater flow results from the case that resulted in the highest intact steam generator steam and feedwater flow rates.

Break Flow Flashing Fraction

A portion of the break flow will flash directly to steam upon entering the secondary side of the ruptured steam generator. Since a transient break flow calculation is not performed for D. C. Cook a detailed time dependent flashing fraction that incorporates the expected changes in primary side temperatures cannot be calculated. Instead a conservative calculation of the flashing fraction is performed using the limiting conditions from the break flow calculation cases. Two time intervals are considered, as in the break flow calculations; pre- and post- reactor trip (SI initiation occurs concurrently with reactor trip). Since the RCS and SG conditions are different before and after the trip, different flashing fractions would be expected.

The flashing fraction is based on the difference between the primary side fluid enthalpy and the saturation enthalpy on the secondary side. Therefore, the highest flashing will be predicted for the case with the highest primary side temperatures. For the flashing fraction calculations it is conservatively assumed that all of the break flow is at the hot leg temperature (the break is assumed to be on the hot leg side of the steam generator). Similarly, a lower secondary side pressure maximizes the difference in the primary and secondary enthalpies resulting in less

flashing. For conservatism in the radiological calculations, it was assumed that reactor trip is assumed at time 0 seconds. Although the pre-trip flashing fraction would be higher due to the lower secondary pressure, the pre-trip flashed break flow passes through the condenser, which reduces the iodine concentration in the steam by a factor of 100. This benefit of the condenser more than offsets the penalty produced by the higher pre-trip flashing fraction. The limiting radiological consequences would therefore be the case with the highest post-trip flashed break flow. All cases consider the same post-trip RCS pressure of 1990 psia and post-trip SG pressure of 888 psia. The highest post-trip flashing fraction, based on the range of operating temperatures covered by this analysis, is for a case with a hot leg temperatures of 615.2°F. It is conservatively assumed that the hot legs' temperature is not reduced for the 30 minutes in which break flow is calculated.

4.1.4 Results of Thermal and Hydraulic Analysis

The limiting tube rupture break flow, the limiting ruptured steam generator atmospheric steam releases from 0 to 30 minutes and limiting flashed break flow are provided in Table 13 along with the long-term steam releases for use in radiological consequences analysis. For an SGTR event, the amount of radioactivity released to the atmosphere is calculated from activity released through the safety valves associated with the ruptured steam generator (from the flashed break flow and steam releases). Therefore, the worst radiological consequences result from the SGTR case with the greatest amount of flashed break flow and steam released. Likewise, a greater break flow results in greater radiological contamination of the secondary side which in turn results in a greater amount of activity released along with the flashing and steaming.

4.1.5 Conclusions of Thermal and Hydraulic Analysis

The SGTR thermal-hydraulic analysis for use in the radiological consequences calculation has been completed in support of NUREG-1465 Dose Program. Based on a primary and secondary side mass and energy balance, the break flow and atmospheric steam releases from the ruptured and intact steam generators were calculated for 30 minutes. After 30 minutes, it was assumed that steam is released only from the intact steam generator in order to dissipate the core decay heat and to subsequently cool the plant down to the RHR system operating conditions. For D. C. Cook, it is conservatively assumed that plant cooldown to RHR operating conditions is not accomplished until 30 days after initiation of the SGTR event and that steam releases are terminated at this time. A primary and secondary side mass and energy balance was used to calculate the steam release and feedwater flow for the intact steam generator from 0 to 2 hours, from 2 to 8 hours, 8 to 24 hours, 1 to 7 days and 7 to 30 days.

4.2 Steam Generator Tube Rupture Radiological Analysis

For the steam generator tube rupture (SGTR), the complete severance of a single steam generator tube is assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive reactor coolant is discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through the main condenser, the atmospheric dump valves, or the safety valves (MSSVs). In addition, iodine activity is contained in the secondary coolant prior to the accident and some of this activity is released to the atmosphere as a result of steaming from the SGs following the accident.

4.2.1 Input Parameters and Assumptions

The analysis of the SGTR radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan Section 15.6.3 (Reference 15) and the draft regulatory guide DG-1081 (Reference 11). For the pre-accident iodine spike case it is assumed that a reactor transient has occurred prior to the SGTR and has raised the RCS iodine

concentration to the Technical Specification limit for a transient of 60 μCi/gm of dose equivalent (DE) I-131 (60 times the Technical Specification coolant equilibrium concentration limit of 1.0 μCi/gm of DE I-131). For the accident-initiated iodine spike case, the reactor trip associated with the SGTR creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131 (Reference 15). The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-cladding gap. Based on having 12 percent of the iodine in the fuel-cladding gap, the gap inventory would be depleted in 6.0 hours and the spike is terminated at that time. The equilibrium spike appearance rate for 1.0 μCi/gm of DE I-131 is assumed to equal the maximum iodine removal rate. The maximum iodine removal rate includes removal by the letdown purification system with a flowrate of 120 gpm, decay of the nuclides and leakage from the primary system of 12 gpm (Technical Specification total primary system leakage).

The noble gas activity concentration in the RCS at the time the accident occurs is based on a one percent fuel defect level. This is approximately equal to the Technical Specification value of 100/E bar μCi/gm for gross radioactivity. The iodine activity concentration of the secondary coolant at the time the SGTR occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131.

The amount of primary to secondary SG tube leakage in the intact SGs is assumed to be equal to the Technical Specification limit for Unit 2 for the total leakage of 1 gpm.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used (Reference 15). Prior to reactor trip and concurrent loss of offsite power an iodine removal factor of 0.01 could be taken for steam released to the condenser, but conservatively, the pre-trip condenser iodine removal is ignored.

All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Break flow flashing fractions and steam release rates from the intact and ruptured steam generators were calculated. The amount of break flow that flashes to steam is conservatively calculated assuming that all break flow is from the hot leg side of the break and that the primary temperatures remain constant.

The SI setpoint will be reached at ~5.6 minutes from event (from the thermal and hydraulic portion of the analysis). The SI signal causes the control room HVAC to switch from the normal operation mode to the accident mode of operation (see Table 3) with a 66 second delay. It is conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 7 minutes after the event initiation.

At 30 days after the accident the RHR System is assumed to be placed into service for heat removal and there are no further steam releases to the atmosphere from the secondary system.

The CEDE dose conversion factors are given in Table 1. The average gamma disintegration energies and noble gas dose conversion factors are given in Table 2. The primary coolant activities used in the dose calculations are given in Table 5 for 1.0 μCi/gm DE I-131 for iodines and 1% fuel defects for noble gases. The pre-accident iodine spike case assumes 60 times the iodine values provided in Table 5. The secondary side activity is 0.1 times the iodine activities provided in Table 5. The parameters associated with the control room HVAC modes are summarized in Table 3. The remaining major assumptions and parameters used specifically in

this analysis are itemized in Table 14. The steam release flowrates were calculated as part of the thermal and hydraulic analysis described in Section 4.1.

4.3 Acceptance Criteria

The criterion for the control room dose are 5 rem TEDE per Draft 10CFR50.67 (Reference 12).

4.4 Results and Conclusions

The control room doses due to the SGTR accident are below.

For the pre-accident iodine spike:	0.4 rem TEDE
For the accident-initiated iodine spike:	0.2 rem TEDE

The acceptance criteria is met.

5.0 LOCKED ROTOR ACCIDENT

An instantaneous seizure of a reactor coolant pump rotor is assumed to occur which rapidly reduces flow through the affected reactor coolant loop. Fuel cladding damage may be predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or safety valves. In addition, iodine activity is contained in the secondary coolant prior to the accident and some of this activity is released to atmosphere as a result of steaming from the steam generators following the accident.

Analyses performed following the guidelines in the Standard Review Plan Section 15.3-3 and 15.3-4 (Reference 20) assume that if the minimum DNBR falls below the limit fuel failure must be assumed. In this case the gap activity of those rods would be released to the reactor coolant system and then be available for release via tube leakage and subsequent steaming.

The draft regulatory guide DG-1081 (Reference 11) states that "although the NRC staff has traditionally relied upon the departure from nuclear boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases".

Thus, for the analysis of radiological consequences utilizing NUREG-1465 source terms, which primarily follows DG-1081, rods with DNBR below the limit are not automatically assumed to fail. Based on a review of fuel failure mechanisms and the conservative predicted behavior of fuel rods during a locked rotor transient, a peak cladding temperature (PCT) criterion of 2700°F is established which is sufficient to preclude fuel rod failures during the locked rotor event. The analysis of the D. C. Cook locked rotor transient confirms that the PCT remains well below this limit. Therefore no fuel rods are assumed to fail as a result of the locked rotor event.

For control room doses, the locked rotor is similar to the loss of offsite power case discussed in Section 10 of this report. The primary and secondary coolant activities are limited by the Technical Specification, as is the primary to secondary leak rate. The atmospheric steam releases from a locked rotor would be less than those calculated for the loss of load and the release point would be the same. The control room dose calculations for the loss of load do not credit isolation of the control room. Therefore it is concluded that the control room dose for the locked rotor is bounded by those calculated in Section 10 for the loss of load.

6.0 ROD EJECTION ACCIDENT

It is assumed that a mechanical failure of a control rod mechanism pressure housing has occurred, resulting in the ejection of a rod cluster control assembly and drive shaft. As a result of the accident, some fuel cladding damage and a small amount of fuel melt are assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive reactor coolant is discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the main condenser or the atmospheric relief valves for the main steam safety valves. Iodine and alkali metals group activity is contained in the secondary coolant prior to the accident, and some of this activity is also released to the atmosphere as a result of steaming the steam generators following the accident. Finally, radioactive reactor coolant is discharged to the containment via the spill from the opening in the reactor vessel head. A portion of this radioactivity is released through containment leakage to the environment.

6.1 Input Parameters and Assumptions

A summary of input parameters and assumptions is provided in Table 15.

The emergency HVAC system is actuated, following a SI-signal for this event. Based on the small break LOCA current licensing basis the SI-signal occurs early in the event (~40 seconds). The emergency HVAC system operation is assumed to start at 2 minutes after event initiation.

6.1.1 Source Term

As a result of the rod ejection accident, less than 15% of the fuel rods in the core undergo DNB. In determining the doses following a rod ejection accident, it is conservatively assumed that 15% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released. Note: Reference 21 calculated 10% rods in DNB which is typically assumed in the radiological consequences, 15% of the rods entering DNB is used for conservatism in this analysis. NUREG-1465 (Reference 1) does not propose a gap release magnitude that should be applied to fission product releases resulting from reactivity insertion accidents such as the rod ejection accident. The non-LOCA gap fractions specified in the draft regulatory guide DG-1081 (Reference 11) are selected for use in the rod ejection analysis to provide the most conservative results. These gap fractions are: 12% for iodine I-131, 15% for Kr-85 and 10% for all other iodines, noble gases and alkali metals.

A small fraction of the fuel in the failed fuel rods is assumed to melt as a result of the rod ejection accident. This amounts to 0.375% of the core and the melting takes place in the centerline of the affected rods. The 0.375% of the fuel assumes that 15% of the rods in the core enter DNB. Of the rods that enter DNB, 50% will experience melting of the fuel (7.5% of the core). Of the percentage of rods experiencing melting, 50% of the axial length of the rod will experience the melting (3.75% of the core). Of the 3.75% only 10% of the radial portion of the rod experiences melting (0.375% of the total core). All activity is released to the RCS coolant from the gap and 35% of the activity from the melted fuel per NUREG-1465 (Reference 1).

A pre-existing iodine spike in the reactor coolant is assumed to have increased the primary coolant iodine concentration to 60 μCi/gm of dose equivalent 1-131 prior to the rod ejection accident. The alkali metals and noble gas activity concentrations in the RCS at the time the accident occurs are based on operation with a fuel defect level of one percent. The iodine activity concentration of the secondary coolant at the time the rod ejection accident occurs is assumed to be equivalent to 0.1 μCi/gm of dose equivalent I-131.

6.1.2 Iodine Chemical Form

NUREG-1465 specifies that the iodine released from the fuel is in the form of 95% cesium iodide, 4.85% elemental, and 0.15% organic. These fractions are used for the containment leakage release pathway. However for the primary to secondary leakage pathway, draft regulatory guide DG-1081 (Reference 11) states that the iodine in solution should be considered to be all elemental and that after iodine is released to the environment the iodine should be modeled as 97% elemental and 3% organic. Since the Donald C. Cook control room filtration system has the same efficiency for elemental and organic iodine, the iodine is modeled as all elemental in the primary to secondary leakage case for simplification.

6.1.3 Release Pathways

Conservatively, all the iodine, alkali metals group and noble gas activity (from prior to the accident and resulting from the accident) is assumed to be in the primary coolant (and not in the containment) when determining doses due to the primary to secondary steam generator tube leakage. The primary to secondary tube leakage and steaming from the steam generators continues until the reactor coolant system pressure drops below the secondary pressure.
A bounding time of 2 hours was selected for this analysis, although the current analyses of record for the small break LOCA peak cladding temperature have shown in the pressure transient curves that this would occur well before that time. A rod ejection pressure transient is similar to that of a small break LOCA.

The primary to secondary steam generator tube leak used in the analysis is a total of 1 gpm consistent with the Technical Specification for Unit 2. Although the primary to secondary pressure differential drops throughout the event, the constant flow rate is maintained.

When determining the doses due to containment leakage, all of the iodine, alkali metal and noble gas activity is assumed to be in the containment. The design basis containment leak rate of 0.25 wt% per day is used for the initial 280 hours. Thereafter, the containment leak rate is assumed to be one-half the design value, or 0.125 wt% per day (Reference 7).

6.1.4 Removal Coefficients

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

An iodine partition factor in the steam generators of 0.01 curies/gm steam per curies/gm water is used (Reference 15). This partition factor is also used for the alkali metal activity in the steam generators.

For the containment leakage pathway, no credit is taken for sedimentation or plateout onto containment surfaces or for containment spray operation which would remove airborne particulates and elemental iodine.

6.2 Acceptance Criteria

The criterion for the control room dose are 5 rem TEDE per Draft 10CFR50.67 (Reference 12).

6.3 Results and Conclusions

The rod ejection accident control room dose is 2.8 rem TEDE

The acceptance criteria is met.

7.0 FUEL HANDLING ACCIDENT

A fuel assembly is assumed to be dropped and damaged during refueling. Analysis of the accident is performed with assumptions selected so that the results are bounding for the accident occurring either inside containment or in the fuel handling building. Activity released from the damaged assembly is released to the outside atmosphere through either the containment purge system or the fuel-handling building ventilation system.

7.1 Input Parameters and Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 16. All of the activity released from the damaged fuel is assumed to be released within two hours (Reference 11). The control room atmospheric dispersion factors for releases from the unit vent were chosen for use in this analysis since this bounds the 0 to 2 hour atmospheric dispersion factor for releases from the containment building surface. The control room HVAC is assumed to be placed into emergency operation at 30 minutes following a fuel handling accident. This assumption assumes manual operator action within 30 minutes to turn on one pressurization fan. No credit is taken for a high radiation monitor signal to initiate emergency HVAC operation.

7.1.1 Source Term

The calculation of the radiological consequences following a fuel handling accident (FHA) uses the gap fractions consistent with the Draft Regulatory Guide DG-1081 (Reference 11) i.e., 12% for I-131, 15% for Kr-85 and 10% for all other isotopes. The gap inventory includes the iodines, noble gases, and the alkali metals (cesium & rubidium).

As in the existing licensing basis, it is assumed that all of the fuel rods in the equivalent of one fuel assembly are damaged to the extent that all their gap activity is released. The assembly inventory is based on the assumption that the subject fuel assembly has been operated at 1.65 times the core average power.

The decay time used in the analysis is 100 hours.

7.1.2 Fission Product Form

While NUREG-1465 specifies that the iodine released from the fuel is in the form of 95% cesium iodide, 4.85% elemental, and 0.15% organic, the analysis assumes that the iodine is 99.75% elemental and 0.25% organic. This is consistent with NRC guidance in draft DG-1081 (Reference 11). In actuality, the nonvolatile cesium iodide would all be retained in the water although gradual conversion of the cesium iodide to form elemental iodine would slowly increase the amount of iodine in the volatile form and which might be released to the environment.

This assumption of 99.75% elemental iodine and 0.25% organic iodine also is consistent with the existing licensing basis analysis and with Regulatory Guide 1.25 (Reference 14).

7.1.3 Pool Scrubbing Removal of Activity

Per the technical specifications, it is assumed that there is a minimum of 23 feet of water above the reactor pressure vessel flange and the spent fuel racks. With this water depth the decontamination factor (DF) of 500 specified by DG-1081 (Reference 11) for elemental iodine would apply. However, in recognition that fuel rod pressure might exceed 1200 psig (but would be less than 1500 psig), the DF is reduced to 400. The DF for organic iodine and noble gases is 1.0.

The cesium released from the damaged fuel rods is assumed to remain in a nonvolatile form and would not be released from the pool.

7.1.4 Isolation and Filtration of Release Paths

No credit is taken for removal of iodine by filters nor is credit taken for isolation of release paths. Although the containment purge will be automatically isolated on a purge line high radiation alarm, isolation is not modeled in the analysis. The activity released from the damaged assembly is assumed to be released to the outside atmosphere over a 2 hour period.

7.2 Acceptance Criteria

The criterion for the control room dose are 5 rem TEDE per Draft 10CFR50.67 (Reference 12).

7.3 Results and Conclusions

The fuel handling accident control room dose is 1.7 rem TEDE

The acceptance criteria is met.

8.0 STEAM LINE BREAL RADIOLOGICAL CONSEQUENCES

The complete severance of a main steam line outside containment is assumed to occur. The affected SG will rapidly depressurize and release radioiodines initially contained in the secondary coolant and primary coolant activity, transferred via SG tube leaks, directly to the outside atmosphere. A portion of the iodine activity initially contained in the intact SGs and noble gas activity due to tube leakage is released to atmosphere through either the atmospheric dump valves (ADV) or the safety valves (MSSVs). The steam line break outside containment will bound any break inside containment since the outside break provides a means for direct release into the environment. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the control room doses resulting from this release.

8.1 Input Parameters and Assumptions

The analysis of the steam line break (SLB) radiological consequences uses the analytical methods and assumptions outlined in the draft regulatory guide DG-1081 (Reference 11). One hundred and two (102) percent of the nominal power level of 3588 MWt (3660 MWt) is used in the analysis. For the pre-accident iodine spike it is assumed that a reactor transient has occurred prior to the SLB and has raised the RCS iodine concentration to 60 times the Technical Specification limit of 1.0 μCi/gm of dose equivalent (DE) I-131. For the accident initiated iodine spike the reactor trip associated with the SLB creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to the maximum equilibrium RCS Technical Specification concentration of 1.0 μCi/gm of DE I-131 (References 11 and 16). Based on having 12 percent of the iodine in the fuel-cladding gap, the gap inventory would be depleted in 6.0 hours and the spike is terminated at that time. The equilibrium spike appearance rate for 1.0 μCi/gm of DE I-131 is assumed to equal the maximum iodine removal rate. The maximum iodine removal rate includes removal by the letdown purification system with a flowrate of 120 gpm, decay of the nuclides and leakage from the primary system of 12 gpm (Technical Specification total primary system leakage).

The noble gas activity concentration in the RCS at the time the accident occurs is based on a fuel defect level of 1.0%. This is approximately equal to the Technical Specification value of 100/E bar μCi/gm for gross radioactivity. The iodine activity concentration of the secondary coolant at the time the SLB occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131.

The amount of primary to secondary SG tube leakage in the faulted SG is assumed to be equal to the Technical Specification limit for Unit 2 for a single SG of 500 gpd. The amount of primary to secondary SG tube leakage in the 3 intact SGs (total) is assumed to be equal to the Technical Specification limit for Unit 2 for the total leakage of 1 gpm minus the faulted SG tube leakage of 500 gpd or a total of 940 gpd to the intact SGs. The tube leakage to both the faulted and intact steam generator persist for 30 days following the initiation of the event.

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power.

The SG connected to the broken steam line is assumed to boil dry within the initial two minutes following the SLB. The entire liquid inventory of this SG is assumed to be steamed off and all of the iodine initially in this SG is released to the environment. Also, iodine carried over to the faulted SG by tube leaks is assumed to be released directly to the environment with no credit taken for iodine retention in the SG. An iodine partition factor in the intact SGs of 0.01 (curies l /gm steam)/(curies l/gm water) is used (Reference 16).

All noble gas activity carried over to the secondary through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Thirty days after the accident, the RHR System is assumed to be placed into service for heat removal and that there are no further steam releases to atmosphere from the secondary system. Steam releases up to 30 days from the intact and faulted SGs have been calculated and provided in Table 17.

It is assumed that the HVAC system is in Normal Mode. On containment isolation, which is conservatively assumed to begin 5 minutes after event initiation, the system is automatically shifted to Accident Mode, where it remains throughout the event.

The emergency HVAC system is actuated, following a SI-signal for this event. Based on a large steam line break event in UFSAR Section 14.2.5, a SI-signal occurs early in the event (within the first minute). The emergency HVAC system operation is conservatively assumed to start at 5 minutes after event initiation.

No fuel failure (DNB or melt) is calculated to occur for the steam line break event.

The major assumptions and parameters used specifically in this analysis are itemized in Table 17.

8.2 Acceptance Criteria

The criterion for the control room dose is 5 rem TEDE per Draft 10CFR50.67 (Reference 12).

8.3 Results and Conclusions

The control room doses due to the steam line break accident are below.

For the pre-accident iodine spike:	0.11 rem TEDE
For the accident-initiated iodine spike:	0.4 rem TEDE

The acceptance criteria is met.

9.0 GAS DECAY TANK RUPTURE RADIOLOGICAL CONSEQUENCES

For the gas decay tank rupture, there is assumed to be a failure that results in the release of the contents of one gas decay tank. For the volume control tank rupture, there is assumed to be a failure that results in the release of the contents of the volume control tank plus the noble gases and the iodines from the letdown flow until the letdown flow path is isolated.

9.1 Input Parameters and Assumptions

The thyroid dose conversion factors are provided in Table 1. The average disintegration energies for whole body and β-skin doses are provided in Table 2. The control room assumptions and parameters are given in Table 3. The major assumptions and parameters used to determine the doses due to a gas decay tank failure or a volume control tank failure are given in Table 18.

The control room HVAC is assumed to be in the normal operating mode for the duration of the event. Actuation of the emergency mode of operation would be initiated by a high radiation signal. Since D. C. Cook does not have a redundant radiation monitor, no credit is taken for the switchover to the emergency mode of the HVAC system for these events.

9.1.1 Gas Decay Tank Rupture

The inventory of gases in the gas decay tank is assumed to be the same as that of the entire RCS with one percent fuel defects, as provided in Table 4.

A failure in the gaseous waste processing system is assumed to result in release of the tank inventory over a period of five minutes.

9.1.2 Volume Control Tank Rupture

The inventory of gases in the volume control tank is based on continuous operation with one percent fuel defects and without any purge of the gas space.

The inventory of iodine in the tank is based on the RCS activity at the pre-existing iodine spike level of 60 μCi/gram dose equivalent I-131 and with 90 percent of the iodine removed by the letdown demineralizer prior to entering the tank.

As a result of the accident, all of the noble gas in the tank and one percent of the iodine in the tank liquid is assumed to be released to the atmosphere over a period of five minutes.

After event initiation, letdown flow to the volume control tank continues at the maximum flow rate of 120 gpm for fifteen minutes by which time the letdown line is assumed to be isolated. The primary coolant noble gas activities used in the volume control tank rupture dose calculations are based on operation with one percent fuel defects and are given in Table 18. The primary coolant iodine activity is conservatively assumed to be at the pre-existing iodine spike level of 60 μCi/gram dose equivalent I-131 and is reduced by 90% by the letdown demineralizer prior to release. All of the noble gas and one percent of the iodine in the letdown flow are assumed to be released to the environment

9.2 Acceptance Criteria

The criterion for the control room dose is 5 rem TEDE per Draft 10CFR50.67 (Reference 12).

9.3 Results and Conclusions

The control room doses due to the waste gas tank rupture accidents are below.

For the gas decay tank rupture:	0.1 rem TEDE
For the volume control tank rupture:	0.4 rem TEDE

The acceptance criteria is met.

10.0 LOSS OF OFFSITE POWER RADIOLOGICAL CONSEQUENCES

A loss of all A.C. power to the plant auxiliaries (i.e., a loss of offsite power) is assumed to occur. Due to primary to secondary tube leakage, radioactive reactor coolant is discharged from the primary into the secondary system. A portion of this radioactivity is assumed to be released to the outside atmosphere through the atmospheric dump/main steam safety valves. In addition, iodine activity is contained in the secondary coolant prior to the accident and some of this activity is released to the atmosphere as a result of steaming of the SGs following the event. The loss of offsite power radiological analysis is performed to bound the condition II events such as the loss of load/turbine trip event. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the doses resulting from this release.

10.1 Input Parameters and Assumptions

The analysis of the loss of offsite power radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan Section 15.6.3 (Reference 15) and the draft regulatory guide DG-1081 (Reference 11) for the SGTR event. The loss of offsite power event is consistent with the locked rotor event without any fuel cladding failures. In DG-1081 (Reference 11), the locked rotor without fuel failure is considered bounded by the SGTR or SLB events. As such the loss of offsite power is analyzed consistently with these events considering both a pre-accident and accident-initiated iodine spike. For the pre-accident iodine spike case it is assumed that a reactor transient has occurred prior to the loss of offsite power and has raised the RCS iodine concentration to the Technical Specification limit for a transient of 60 μCi/gm of dose equivalent (DE) I-131 (60 times the Technical Specification coolant equilibrium concentration limit of 1.0 μCi/gm of DE I-131). For the accident-initiated iodine spike case, the reactor trip associated with the loss of offsite power creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-cladding gap. Based on having 12 percent of the iodine in the fuel-cladding gap, the gap inventory would be depleted in 6.0 hours and the spike is terminated at that time. The equilibrium spike appearance rate for 1.0 μCi/gm of DE I-131 is assumed to equal the maximum iodine removal rate. The maximum iodine removal rate includes removal by the letdown purification system with a flowrate of 120 gpm, decay of the nuclides and leakage from the primary system of 12 gpm (Technical Specification total primary system leakage).

The noble gas activity concentration in the RCS at the time the accident occurs is based on a one percent fuel defect level. This is approximately equal to the Technical Specification value of 100/E bar μCi/gm for gross radioactivity. The iodine activity concentration of the secondary coolant at the time the SGTR occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131.

The amount of primary to secondary SG tube leakage in the SGs is assumed to be equal to the Technical Specification limit for Unit 2 for the total leakage of 1 gpm.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used (Reference 15). Prior to reactor trip and concurrent loss of offsite power an iodine removal factor of 0.01 could be taken for steam released to the condenser, but conservatively, the pre-trip condenser iodine removal is ignored.

All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Steam release rates from the steam generators were calculated for the duration of the event.

The loss of offsite power would not be expected to result in a Safety Injection signal, therefore the control room HVAC to switch from the normal operation mode to the accident mode of operation would need to occur off a high radiation monitor signal. Since the high radiation monitor is not considered "safety grade" this system is not assumed to operate in this analysis and the control room HVAC system is never assumed to switch to the emergency mode. The analysis only assumes that the control room HVAC system is in the normal mode with 1000 cfm of unfiltered makeup flow into the control room for the duration of the event.

At 30 days after the accident the RHR System is assumed to be placed into service for heat removal and there are no further steam releases to the atmosphere from the secondary system.

The major assumptions and parameters used specifically in this analysis are itemized in Table 19. Steam release rates have been calculated as part of this program and are provided in Table 19.

10.2 Acceptance Criteria

The criterion for the control room dose are 5 rem TEDE per Draft 10CFR50.67 (Reference 12).

10.3 Results and Conclusions

The control room doses due to the loss of offsite power event are below.

For the pre-accident iodine spike:	0.4 rem TEDE
For the accident-initiated iodine spike:	2.0 rem TEDE

The acceptance criteria is met.

11.0 CONCLUSIONS

All accident radiological consequences meet the acceptance criteria. Therefore the control room design is acceptable. The analyses considered a maximum of 2 hours from the start of an event, as the limiting time for the operators to turn off the second pressurization fan. The analyses considered minimum and maximum control room volume assumptions.

12.0 REFERENCES

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20. NUREG-0800, Standard Review Plan 15.3-3 – 15.3-4, "Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break", Rev. 2, July 1981.

21. WCAP-7588, Revision 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," January 1975.

TABLE 1
COMMITTED EFFECTIVE DOSE EQUIVALENT DOSE CONVERSION FACTORS

Isotope	DCF (rem/curie)	Isotope	DCF (rem/curie)
I-131	3.29E4	Cs-134	4.62E4
I-132	3.81E2	Cs-136	7.33E3
I-133	5.85E3	Cs-137	3.19E4
I-134	1.31E2	Rb-86	6.63E3
I-135	1.23E3		
		Ru-103	8.95E3
Kr-85m	N/A	Ru-105	4.55E2
Kr-85	N/A	Ru-106	4.77E5
Kr-87	N/A	Rh-105	9.56E2
Kr-88	N/A	Mo-99	3.960E3
Xe-131m	N/A	Tc-99m	3.3E1
Xe-133m	N/A		
Xe-133	N/A	Y-90	8.44E3
Xe-135m	N/A	Y-91	4.89E4
Xe-135	N/A	Y-92	7.80E2
Xe-138	N/A	Y-93	2.15E3
		Nb-95	5.81E3
Te-127	3.18E2	Zr-95	2.37E4
Te-127m	2.15E4	Zr-97	4.33E3
Te-129m	2.39E4	La-140	4.85E3
Te-129	9.0E1	La-141	5.81E2
Te-131m	6.4E3	La-142	2.53E2
Te-132	9.44E3	Nd-147	6.85E3
Sb-127	6.04E3	Pr-143	1.09E4
Sb-129	6.44E2	Am-241	4.44E8
		Cm-242	1.73E7
Ce-141	8.96E3	Cm-244	2.48E8
Ce-143	3.39E3		
Ce-144	3.74E5	Sr-89	4.14E4
Pu-238	3.92E8	Sr-90	1.3E6
Pu-239	4.3E8	Sr-91	1.66E3
Pu-240	4.3E8	Sr-92	8.1E2
Pu-241	8.26E6	Ba-139	1.7E2
Np-239	2.51E3	Ba-140	3.74E3

TABLE 2
AVERAGE GAMMA DISINTEGRATION ENERGIES AND NOBLE GAS DOSE CONVERSION FACTORS

Isotope	Energy (mev/dis)	Isotope	Energy (mev/dis)
I-131	0.38	Ru-103	0.468
I-132	2.2	Ru-105	0.775
I-133	0.6	Ru-106	0.0
I-134	2.6	Rh-105	0.078
I-135	1.4	Mo-99	0.15
		Tc-99m	0.126
Te-127	4.86E-3		
Te-127m	0.0112	Y-90	1.7E-6
Te-129m	0.0375	Y-91	3.61E-3
Te-129	0.0591	Y-92	0.251
Te-131m	1.42	Y-93	0.0889
Te-132	0.233	Nb-95	0.766
Sb-127	0.688	Zr-95	0.739
Sb-129	1.44	Zr-97	0.179
		La-140	2.31
Ce-141	0.076	La-141	0.0427
Ce-143	0.282	La-142	2.68
Ce-144	0.021	Nd-147	0.14
Pu-238	1.81E-3	Pr-143	8.9E-9
Pu-239	8.08E-4	Am-241	0.0324
Pu-240	1.73E-3	Cm-242	1.83E-3
Pu-241	2.54E-6	Cm-244	1.7E-3
Np-239	0.172		
		Sr-89	8.45E-5
Cs-134	1.55	Sr-90	0.0
Cs-136	2.16	Sr-91	0.693
Cs-137	0.564	Sr-92	1.34
Rb-86	0.0945	Ba-139	0.043
		Ba-140	0.182

Isotope	DCF (rem· m^3 /Ci · sec)	Isotope	DCF (rem· m^3 /Ci · sec)
Kr-85m	0.03707	Xe-133m	0.00553
Kr-85	0.000484	Xe-133	0.00624
Kr-87	0.146	Xe-135m	0.0775
Kr-88	0.37	Xe-135	0.0482
Xe-131m	0.00152	Xe-138	0.198

TABLE 3
CONTROL ROOM PARAMETERS

Breathing Rate - Duration of the Event	3.47E-4 m^3/sec
Volume - Maximum	89890 ft^3
Volume - Minimum	50616 ft^3
Occupancy Factors	
0 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4
Normal Ventilation Flow Rates	
Filtered Makeup Flow Rate	0.0 SCFM
Filtered Recirculation Flow Rate	0.0 SCFM
Max Unfiltered Makeup Flow Rate	1000. SCFM
Unfiltered Recirculation Flow Rate	13400 SCFM
(Not modeled - no impact on analyses)	
Emergency Ventilation System Flow Rates (One Fan)	
Max Filtered Makeup Air Flow Rate	1000 SCFM
Min Filtered Recirculation Flow Rate	4400 SCFM
Max Unfiltered Inleakage	98 SCFM
Unfiltered Recirculation Flow Rate	13400
(Not modeled - no impact on analyses)	
Emergency Ventilation System Flow Rates (Two Fans)	
Max Filtered Makeup Air Flow Rate	2000 SCFM
Min Filtered Recirculation Flow Rate	8800 SCFM
Filter Efficiencies (One Fan)	
Elemental	95%
Organic	95%
Particulate	98%
Filter Efficiencies (Two Fan)	
Elemental	80%
Organic	80%
Particulate	98%
Delay to Switchover of HVAC from Normal Operation to Emergency Operation after Receipt of a S-signal	66 seconds
Time to Turn Off Second Pressurization Fan (Limiting Time is Assumed)	0/30 min/2 hours
TEDE Dose Acceptance Criteria for Control Room	5.0 rem

TABLE 4
CORE TOTAL FISSION PRODUCT ACTIVITIES

BASED ON 102% of 3588 MWT

Isotope	Activity (Ci)	Isotope	Activity (Ci)
I-131	1.02E8	Cs-134	1.88E7
I-132	1.49E8	Cs-136	5.27E6
I-133	2.08E8	Cs-137	9.24E6
I-134	2.24E8	Rb-86	1.70E5
I-135	1.93E8		
		Ru-103	1.63E8
Kr-85m	2.66E7	Ru-105	1.11E8
Kr-85	8.42E5	Ru-106	4.67E7
Kr-87	4.88E7	Rh-105	9.89E7
Kr-88	6.94E7	Mo-99	1.88E8
Xe-131m	7.21E5	Tc-99m	1.62E8
Xe-133m	2.98E7		
Xe-133	2.01E8	Y-90	7.12E6
Xe-135m	4.16E7	Y-91	1.26E8
Xe-135	4.32E7	Y-92	1.30E8
Xe-138	1.67E8	Y-93	1.50E8
		Nb-95	1.74E8
Te-127	9.79E6	Zr-95	1.73E8
Te-127m	1.42E6	Zr-97	1.74E8
Te-129m	8.63E6	La-140	1.89E8
Te-129	3.17E7	La-141	1.70E8
Te-131m	1.57E7	La-142	1.64E8
Te-132	1.46E8	Nd-147	6.69E7
Sb-127	9.98E6	Pr-143	1.53E8
Sb-129	3.36E7	Am-241	5.76E3
		Cm-242	2.42E6
Ce-141	1.71E8	Cm-244	2.69E5
Ce-143	1.57E8		
Ce-144	1.17E8	Sr-89	9.72E7
Pu-238	2.48E5	Sr-90	6.79E6
Pu-239	3.02E4	Sr-91	1.20E8
Pu-240	3.75E4	Sr-92	1.29E8
Pu-241	8.38E6	Ba-139	1.83E8
Np-239	2.13E9	Ba-140	1.81E8

TABLE 5
RCS COOLANT CONCENTRATIONS

BASED ON 1.0 μCi/gm DE I-131 FOR IODINES

AND 1% FUEL DEFECTS FOR NOBLE GASES

Isotope	Activity (μCi/gm)
I-131	0.74111
I-132	0.71511
I-133	1.39641
I-134	0.15342
I-135	0.62409
Kr-85m	1.928
Kr-85	9.401
Kr-87	1.201
Kr-88	4.166
Xe-131m	2.010
Xe-133m	16.40
Xe-133	251.5
Xe-135m	0.5641
Xe-135	7.766
Xe-138	0.6964

Iodine concentrations are converted from the Reference 9 values using the dose conversion factors in ICRP-30 (Reference 4) for direct thyroid doses. Noble gas concentrations are multiplied by 110% to account for 102% of 3600 MWt.

TABLE 6
NUCLIDE DECAY CONSTANTS

Isotope	Decay Constant (hr^{-1})	Isotope	Decay Constant (hr^{-1})
I-131	0.00359	Cs-134	3.84E-5
I-132	0.303	Cs-136	2.2E-3
I-133	0.0333	Cs-137	2.64E-6
I-134	0.791	Rb-86	1.55E-3
I-135	0.105		
		Ru-103	7.35E-4
Kr-85m	0.155	Ru-105	0.156
Kr-85	7.37E-6	Ru-106	7.84E-5
Kr-87	0.547	Rh-105	1.96E-2
Kr-88	0.248	Mo-99	1.05E-2
Xe-131m	0.00241	Tc-99m	0.115
Xe-133m	0.0130		
Xe-133	0.00546	Y-90	1.08E-2
Xe-135m	2.72	Y-91	4.94E-4
Xe-135	0.0756	Y-92	0.196
Xe-138	2.93	Y-93	0.0686
		Nb-95	8.22E-4
Te-127	7.41E-2	Zr-95	4.51E-4
Te-127m	2.65E-4	Zr-97	4.1E-2
Te-129m	8.6E-4	La-140	1.72E-2
Te-129	0.598	La-141	0.176
Te-131m	2.31E-2	La-142	0.45
Te-132	8.86E-3	Nd-147	2.63E-3
Sb-127	7.5E-3	Pr-143	2.13E-3
Sb-129	0.16	Am-241	1.83E-7
		Cm-242	1.77E-4
Ce-141	8.89E-4	Cm-244	4.37E-6
Ce-143	0.021		
Ce-144	1.02E-4	Sr-89	5.72E-4
Pu-238	9.02E-7	Sr-90	2.72E-6
Pu-239	3.29E-9	Sr-91	0.073
Pu-240	1.21E-8	Sr-92	0.256
Pu-241	5.5E-6	Ba-139	0.502
Np-239	0.0123	Ba-140	2.27E-3

TABLE 7
IODINE CHEMICAL SPECIES

Iodine Form	TID-14844	NUREG-1465
Elemental	91%	4.85%
Organic	4%	0.15%
Particulate	5%	95%

TABLE 8
FISSION PRODUCT RELEASE TIMING

Release Phase	Duration (TID-14844)	Duration (NUREG-1465)[1]
Coolant Activity	instantaneous release	10 to 30 seconds
Gap Activity	instantaneous release	0.5 hour
Early In-vessel	instantaneous release	1.3 hour
Ex-vessel	not defined [2]	2 hours [3]
Late In-vessel	not defined [2]	10 hours [3]

1. Releases are sequential with the exception of the ex-vessel and the late in-vessel phases which both being at the end of the early in-vessel release phase.
2. Ex-vessel and late in-vessel release not defined in TID-14844.
3. Per SECY-94-302 (Reference 3), ex-vessel and late in-vessel releases are not applicable to design basis analyses.

TABLE 9
CORE FISSION PRODUCT RELEASE FRACTIONS

	Gap Release [1]		Early In-Vessel	
	TID	NUREG	TID	NUREG
Noble gases	n/a[2]	0.05[3]	1.0	0.95
Halogens	n/a[2]	0.05[3]	0.5[4]	0.35
Alkali Metals	n/a	0.05[3]	0.01[5]	0.25
Tellurium group	n/a	0	0.01[5]	0.05
Barium, Strontium	n/a	0	0.01[5]	0.02
Noble Metals (Ruthenium group)	n/a	0	0.01[5]	0.0025
Cerium group	n/a	0	0.01[5]	0.0005
Lanthanides	n/a	0	0.01[5]	0.0002

(1) The TID-14844 methodology does not specifically address the gap release. The NUREG-1465 methodology assumes that gap and early in-vessel (core melt) releases are sequential. The TID-14844 source term model assumes the instantaneous release of 50% of core iodine and noble gases, with no distinction made between gap activity release and early in-vessel release. The NUREG-1465 source term assumes a release of gap activity (5% of core) followed by 35% in-vessel release for a total release of 40% of core.

(2) Gap fraction is not defined by TID-14844.

(3) 3% immediate release at 30 seconds followed by 2% over 0.5 hours

(4) Per TID-14844, half of this is assumed to plate out instantaneously.

(5) Referred to in TID-14844 as "other fission products" but not typically included in dose analyses.

TABLE 10
NUREG-1465 NUCLIDE GROUPS

Group	Title	Elements in Group
1	Noble Gases	Xe, Kr
2	Halogens	I, Br
3	Alkali Metals	Cs, Rb
4	Tellurium Group	Te, Sb, Se
5	Barium, Strontium	Ba, Sr
6	Noble Metals	Ru, Rh, Pd, Mo, Tc, Co
7	Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
8	Cerium Group	Ce, Pu, Np

TABLE 11
ASSUMPTIONS USED FOR LARGE BREAK LOCA ANALYSIS

Source Term

Core Activity	See Table 4
Activity release fractions and timing	See Table 9
Iodine chemical form in containment	
Elemental	4.85%
Organic	0.15%
Particulate (cesium iodide)	95%
Iodine chemical form in sump & recirculating liquid	100% elemental

Containment

Containment Volume (Rounded up values, annular region increased by 10%, used in Radiological Consequences Calculation, Spray Removal Coefficients Based on Values in Sections 2.2.3.1 and 2.2.3.2)

Upper	9.0E5 ft^3
Active Lower	3.0E5 ft^3
Annular	6.9E4 ft^3

Flow Rates	
Between Active Lower to Upper	
0 – 3 minutes	0.0 cfm
after 3 minutes	39000 cfm
Between Active Lower and Annular	500 cfm
Leak Rate	
0 - 280 hours	0.25 wt% per day
>280 hours	0.125 wt% per day
Duration of releases	30 days
Spray Operation	
Time to initiate sprays	126 seconds
Time to initiate switchover to recirculation spray operation	1.25 hours
Time required to switchover to recirculation spray operation	5 minutes
Time to start RHR Spray	1.333 hours
Time terminate recirculation and RHR spray	6 hours
Spray Flow Rates	
Upper	1950 gpm
Active Lower	700 gpm
RHR Spray	1850 gpm
Spray Fall Height	
Upper	59 feet
Active Lower	26 ft
RHR Spray	41 ft

TABLE 11
ASSUMPTIONS USED FOR LARGE BREAK LOCA ANALYSIS
(continued)

Spray Removal Coefficients (reduced by 10% from calculated values)	
Elemental iodine (injection spray phase)	
Upper	20.0 hr^{-1}
Active Lower	8.9 hr^{-1}
Elemental iodine (recirculation spray phase)	
Upper	7.8 hr^{-1}
Active Lower	3.3 hr^{-1}
Particulates (until DF = 50)	
Upper	3.8 hr^{-1}
Upper Including RHR	6.3 hr^{-1}
Active Lower	1.8 hr^{-1}
Elemental iodine DF limit	200
Credited Sump Volume	
0 – 2 hours	2.1E5 gal
2 – 8 hours	3.29E5 gal
8 – 24 hours	3.53E5 gal
24 – 48 hours	3.77E5 gal
48 – 100 hours	4.01E5 gal
100 – 150 hours	4.31E5 gal
150 – 200 hours	4.61E5 gal
200 – 250 hours	4.79E5 gal
250 – 300 hours	4.91E5 gal
300 – 336 hours	5.03E5 gal
after 336 hours	5.1E5 gal
ECCS Leak Rate (0 – 30 days)	0.2 gpm
Passive Failure Leak Rate (24 – 24.5 hours)	50 gpm
Purge Flow Rate	16000 cfm
Purge Isolation	10 seconds

TABLE 11
ASSUMPTIONS USED FOR LARGE BREAK LOCA ANALYSIS
(continued)

Control Room Parameters	See Table 3
Atmospheric Dispersion Factors	
Control room for containment leakage	
0-2 hr	8.95E-4 sec/m^3
2 - 8 hr	6.91E-4 sec/m^3
8 - 24 hr	2.79E-4 sec/m^3
24 - 96 hr	2.23E-4 sec/m^3
96 - 720 hr	2.63E-4 sec/m^3
Control room for purge release, ECCS leakage and passive failure	
0-2 hr	1.74E-3 sec/m^3
2 - 8 hr	1.00E-3 sec/m^3
8 - 24 hr	4.19E-4 sec/m^3
24 - 96 hr	3.09E-4 sec/m^3
96 - 720 hr	2.69E-4 sec/m^3

TABLE 12
ASSUMPTIONS USED FOR SMALL BREAK LOCA ANALYSIS

Source Term	
Core activity	See Table 4
Fission product gap fractions	
Iodines	5%
Noble gases	5%
Cesium & Rubidium	5%
Fraction of fuel rods in core failing	100%
Iodine Chemical Form (containment release path)	
Elemental	4.85%
Organic	0.15%
Particulate	95%
Iodine Chemical Form (steam generator steaming path)	
Iodine chemical form	100% elemental
Containment Leakage Release Path	
Sedimentation particulate removal coefficient	0.1 hr^{-1}
DF limit for particulates (not reached in 65 hours)	1000
Leak rate	
0 - 280 hours	0.25 wt% per day
>280 hours	0.125 wt% per day
Duration of releases	30 days
Steam Generator Steaming Release Path	
Primary coolant mass	2.41E8 g
Secondary coolant mass (total of 4 SGs)	1.65E8 g
Primary to Secondary leak rate	3800 g/min
Steaming rate from the secondary side	1.75E6 g/min
Steaming partition coefficient	0.01
Duration of releases	7200 seconds

TABLE 12
ASSUMPTIONS USED FOR SMALL BREAK LOCA ANALYSIS
(continued)

Control Room Parameters	See Table 3
Atmospheric Dispersion Factors	
Control room (containment building release)	
0-2 hr	8.95E-4 sec/m^3
2 - 8 hr	6.91E-4 sec/m^3
8 - 24 hr	2.79E-4 sec/m^3
24 - 96 hr	2.23E-4 sec/m^3
>96 hr	2.63E-4 sec/m^3
Control room (steam generator PORV release)	
0-2 hr	5.01E-4 sec/m^3

TABLE 13
BOUNDING SGTR THERMAL-HYDRAULIC RESULTS FOR RADIOLOGICAL DOSE ANALYSIS

Post Trip (after 0 seconds)	
Tube Rupture Break Flow	162,000 lbm
Percentage of Break Flow Which Flashes	16.77%
Steam Release from Ruptured SG up to 2 Hours	73,000 lbm
Steam Release from Intact SGs up to 2 Hours	565,000 lbm
Steam Release from Intact SGs for 2 - 8 Hours	1,505,000 lbm
Steam Release from Intact SGs for 8 - 24 Hours	1,482,000 lbm
Steam Release from Intact SGs for 1 - 7 Days	9,729,000 lbm
Steam Release from Intact SGs for 7 - 30 Days	36,871,000 lbm
Time at Which SI Setpoint is Reached	5.6 minutes

TABLE 14
ASSUMPTIONS USED FOR SGTR DOSE ANALYSIS

Power (102%)	3660 MWt
Reactor Coolant Noble Gas Activity Prior to accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	
Pre-Accident Spike	60 μCi/gm of DE I-131
Accident Initiated Spike	1.0 μCi/gm of DE I-131
Reactor Coolant Iodine Appearance Rate Increase Due to the Accident-Initiated Spike	500 times equilibrium rate
Duration of Accident-Initiated Spike	6.0 hours
Secondary Coolant Activity Prior to Accident	0.1 μCi/gm of DE I-131
SG Tube Leak Rate for Intact SGs During Accident	1 gpm
Time of Reactor Trip	0 sec - assumed for flashing flow 335 sec – assumed for Emergency HVAC
Break Flow to Ruptured SG	See Table 13
Break Flow Flashing Fraction	See Table 13
Steam Release from SGs to Environment	See Table 13
Pre-Trip Condenser Iodine Removal Factor	N/A
SG Iodine Partition Factor	0.01
Duration of Activity Release from Secondary System	30 days
Offsite Power	Lost
Initiation of the Emergency HVAC Operation	7 min
<u>Control Room Parameters</u>	Table 3
<u>Atmospheric Dispersion Factors</u>	
Control room (steam generator steaming pathway)	
0 - 2 Hours	5.01E-4 sec/m^3
2 - 8 Hours	3.16E-4 sec/m^3
8 - 24 Hours	1.35E-4 sec/m^3
1 - 4 days	1.06E-4 sec/m^3
4 - 30 days	7.51E-5 sec/m^3

TABLE 15
ASSUMPTIONS USED FOR ROD EJECTION ACCIDENT

Source Term

Core activity	See Table 4
Fission product gap fractions	
I-131	12%
Other Iodines	10%
Kr-85	15%
Other Noble gases	10%
Cesium & Rubidium	10%
Fraction of fuel rods in core failing	15%
Fraction of fuel melting	0.375%
Activity release from melted fuel	
Iodines	35%
Noble gases	95%
Cesium & Rubidium	25%

Primary coolant activity before fuel failure
- includes pre-existing iodine spike (Ci)

I-131	1.07E4
I-132	1.03E4
I-133	2.02E4
I-134	2.22E3
I-135	9.02E3
Cs-134	6.85E2
Cs-136	7.09E2
Cs-137	4.20E2
Rb-86	1.96E3
Kr-85m	4.23E2
Kr-85	2.06E3
Kr-87	2.63E2
Kr-88	9.13E2
Xe-131m	4.40E2
Xe-133m	3.59E3
Xe-133	5.51E4
Xe-135m	1.24E2
Xe-135	1.70E3
Xe-138	1.53E2

Secondary coolant activity at beginning of event (Ci)

I-131	15.4
I-132	15.0
I-133	29.2
I-134	3.1
I-135	12.9

TABLE 15
ASSUMPTIONS USED FOR ROD EJECTION ACCIDENT
(continued)

Iodine Chemical Form (containment release path)	
Elemental	4.85%
Organic	0.15%
Particulate	95%
Iodine Chemical Form (steam generator steaming path)	
Iodine chemical form	100% elemental
Containment Leakage Release Path	
Removal coefficients	None assumed
Leak rate	
0 - 280 hours	0.25 wt% per day
>280 hours	0.125 wt% per day
Duration of releases	30 days
Steam Generator Steaming Release Path	
Primary coolant mass	2.41E8 g
Secondary coolant mass	2.087E8 g
Primary to Secondary leak rate	3800 g/min
Steaming rate from the secondary side	1.739E6 g/min
Steaming partition coefficient	0.01
Duration of releases	7200 seconds
Control Room Parameters	See Table 3
Atmospheric Dispersion Factors	
Control room (containment leakage pathway)	
0-2 hr	8.95E-4 sec/m^3
2 - 8 hr	6.91E-4 sec/m^3
8 - 24 hr	2.79E-4 sec/m^3
24 - 96 hr	2.23E-4 sec/m^3
>96 hr	2.63E-4 sec/m^3
Control room (steam generator steaming pathway)	
0-2 hr	5.01E-4 sec/m^3

TABLE 16
ASSUMPTIONS USED FOR FUEL HANDLING ACCIDENT ANALYSIS

Number of fuel assemblies in core	193
Radial peaking factor	1.65
Fuel rod gap fraction	
I-131	12%
Other iodines	10%
Kr-85	15%
Other noble gases	10%
Fuel damaged	one assembly
Iodine species	
Elemental	99.75%
Organic	0.25%
Particulate	none
Water depth	23 feet
Pool scrubbing factor	
Elemental iodine	400
Organic iodine	1
Noble gases	1
Filter efficiency	No filtration assumed
Release termination	No isolation of release paths assumed
Operator Action to Switch to Control Room Emergency HVAC Operation	30 minutes
Control Room Parameters	See Table 3
Atmospheric Dispersion Factors	
Control Room (0-2 hr)	
Release from unit vent	1.74E-3 sec/m^3
Release from containment	8.95E-4 sec/m^3

TABLE 17
ASSUMPTIONS USED FOR STEAM LINE BREAK DOSE ANALYSIS

Power (102%)	3660 MWt
Reactor Coolant Noble Gas Activity Prior to accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	
Pre-Accident Spike	60 µCi/gm of DE I-131
Accident Initiated Spike	1.0 µCi/gm of DE I-131
Reactor Coolant Iodine Appearance Rate Increase Due to the Accident-Initiated Spike	500 times equilibrium rate
Duration of Accident-Initiated Spike	6.0 hours
Secondary Coolant Activity Prior to Accident	0.1 µCi/gm of DE I-131
Faulted SG Tube Leak Rate During Accident	500 gpd
Intact SGs Tube Leak Rate During Accident	940 gpd total for 3 SGs
SG Iodine Partition Factor	
Intact SG	0.01
Faulted SG	1.0
Duration of Activity Release from Secondary System	30 days
Offsite Power	Lost
Steam Release from Intact SGs to Environment	
0-2 hours	456,000 lbm
2-8 hours	1,186,000 lbm
8-24 hours	1,347,000 lbm
1-7 days	8,844,000 lbm
7 -30 days	33,519,000 lbm
Steam Release from Faulted SG to Environment during First Two Minutes	161,000 lbm (activity corresponding to this mass is released from the faulted SG)
Time to Switch Control Room HVAC to Accident Mode, min.	5
<u>Control Room Parameters</u>	Table 3

TABLE 17
ASSUMPTIONS USED FOR STEAM LINE BREAK DOSE ANALYSIS
(continued)

Atmospheric Dispersion Factors

Control Room Atmospheric Dispersion Factors from PORVs to Control Room Intake for Intact SGs Releases

0 - 2 Hours	5.01E-4 sec/m^3
2 - 8 Hours	3.16E-4 sec/m^3
8 - 24 Hours	1.35E-4 sec/m^3
1 - 4 days	1.06E-4 sec/m^3
4 - 30 days	7.51E-5 sec/m^3

Control Room Atmospheric Dispersion Factors from Unit Vent to Control Room Intake for Faulted SG Releases (bounds releases from containment building)

0 - 2 Hours	1.74E-3 sec/m^3
2 - 8 Hours	1.00E-3 sec/m^3
8 - 24 Hours	4.19E-4 sec/m^3
1 - 4 days	3.09E-4 sec/m^3
4 - 30 days	2.69E-4 sec/m^3

TABLE 18
ASSUMPTIONS USED FOR GAS DECAY TANK AND VOLUME CONTROL TANK RUPTURE DOSE ANALYSIS

Gas Decay Tank Failure Assumptions

Gas Decay Tank Inventory (Ci)	
Kr-85m	464.6
Kr-85	2265.6
Kr-87	289.4
Kr-88	1004.
Xe-131m	484.4
Xe-133m	3952.4
Xe-133	60611.5
Xe-135m	135.9
Xe-135	1871.6
Xe-138	167.8
Duration of release	5 minutes

Volume Control Tank Failure Assumptions

Volume Control Tank Gas Inventory (Ci)	
Kr-85m	141.4
Kr-85	1330.6
Kr-87	28.3
Kr-88	470.3
Xe-131m	241.1
Xe-133m	1829.5
Xe-133	30850.
Xe-135m	69.
Xe-135	767.9
Xe-138	4.3
Volume Control Tank Liquid Iodine Inventory (Ci) - Includes Demineralizer DF of 10	
I-131	1072.
I-132	1034.
I-133	2019.
I-134	221.8
I-135	.902.3
Duration of release	5 minutes
Reactor Coolant Activity	
Iodines	60 μCi/g of DE I-131
Noble Gases	1.0% fuel defect level
Letdown Flow Rate	120 gpm
Letdown Demineralizer Decontamination Factor for Iodine	10
Letdown Flow Isolation Time	15 minutes

Common Assumptions

Control Room Parameters	See Table 3
Atmospheric Dispersion Factors	
Control Room (0-2 hr) Release from unit vent	1.74E-3 sec/m^3

TABLE 19
ASSUMPTIONS USED FOR LOSS OF OFFSITE POWER DOSE ANALYSIS

Power (102%)	3660 MWt
Reactor Coolant Noble Gas Activity Prior to accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	
Pre-Accident Spike	60 μCi/gm of DE I-131
Accident Initiated Spike	1.0 μCi/gm of DE I-131
Reactor Coolant Iodine Appearance Rate Increase Due to the Accident-Initiated Spike	500 times equilibrium rate
Duration of Accident-Initiated Spike	6.0 hours
Secondary Coolant Activity Prior to Accident	0.1 μCi/gm of DE I-131
SG Tube Leak Rate for SGs During Accident	1 gpm
Pre-Trip Condenser Iodine Removal Factor	N/A
SG Iodine Partition Factor	0.01
Duration of Activity Release from Secondary System	30 days
Offsite Power	Lost
Steam Release from SGs to Environment	738,000 lbm (0 - 2 hr) 1,345,000 lbm (2 - 8 hr) 1,590,000 lbm (8 - 24 hr) 11,030,000 lbm (1-7 days) 36,871,000 lbm (7 - 30 days)
Initiation of the Emergency HVAC Operation	Never
Control Room Parameters	Table 3
Atmospheric Dispersion Factors	
Control room (steam generator PORV release)	
0 - 2 Hours	5.01E-4 sec/m^3
2 - 8 Hours	3.16E-4 sec/m^3
8 - 24 Hours	1.36E-4 sec/m^3
1 - 4 days	1.06E-4 sec/m^3
4 - 30 days	7.52E-5 sec/m^3

ATTACHMENT B to AEP-00-072

Design Input for D. C. Cook Offsite and Control Room Dose
Analysis Using the Alternate Source Term (DIT-B-00069-06, February 3, 2000)

American Electric Power
Cook Nuclear Plant
One Cook Place
Bridgman, MI 49106
616 465 5901

AMERICAN ELECTRIC POWER

Mr. Don Peck
Westinghouse Electric Corporation
Energy Center
P.O. Box 355
Pittsburgh, PA 15230

February 3, 2000

SUBJECT Design Input for D. C. Cook
Offsite and Control Room Dose Analysis Using the Alternative Source Term

ATTN Jill Monahan

The enclosed document, DIT-B-00069-06, provides the design input for the offsite and control room dose analyses in support of the D. C. Cook Nuclear Plant. The analysis uses the Alternative Source Term.

Revision 6 documents parameters determined by Westinghouse by reference to the revised final owner accepted NUREG-1465 engineering report, AEP-99-487. Revision 6 also corrects errors in the containment spray flow rate (parameters L7 and L8), containment spray fall heights (parameter L10), steam release for the loss of offsite power event (parameter T6), and the time to start the HVAC system for the RCCA ejection accident (parameter R15). Only the change in spray fall heights impacts the analysis reported in AEP-99-487.

If you have any questions concerning these input assumptions, please call me at (616) 697-5106, or Rick Kohrt at (616) 697-5668.

Sincerely,

Jeb Kingseed
Manager, Nuclear Fuels Safety & Analysis

/rjk
cc: R. J. Kohrt
DCN-N-6452.7

AEP DESIGN INFORMATION TRANSMITTALS (DIT)

DIT Form, Part 1

☒ **SAFETY-RELATED** ☐ **NON-SAFETY-RELATED**	Originating Organization ☒ AEP ☐ Other (specify)	DIT No DIT-B-00069-06
D.C. Cook Unit (Circle applicable): ☐ 1 ☐ 2 ☒ **BOTH** System Designation: CRMVT		Page 1 of 30 To Jill Monahan, Westinghouse

Subject: Design Input for D. C. Cook Offsite and Control Room Dose Analysis

Rick Kohrt Preparer	Engineer Position	Rick Kohrt Preparer's Signature	Feb 3, 2000 Date
Alan Meinershagen Reviewer	Project Engineer Position	Alan Meinershagen Reviewer's Signature	Feb 3, 2000 Date
JES KINGSEED Approver	ASST. DIRECTOR Position	Approver's Signature	Date 2/3/2000

Status of Information: ☐ Approved for Use ☒ Unverified

Method and Schedule of Verification for Unverified DITs The calculation for atmospheric dispersion factor must be verified prior to submittal of offsite dose analysis results scheduled for February, 8, 2000.
CR # CR-99-19039

Holds Associated with Unverified DITs: The specific input parameter requiring verification is item C6. Atmospheric dispersion factor for releases from the containment building surface to the site boundary and low population zone is only required for the offsite dose analyses.

Description of Information: See attached table of input parameters.

Revision 6 of the DIT provides changes to input parameters as identified by margin bars.

Purpose of Issuance (Including any Precautions or Limitations): The purpose of issuance is to calculate the radiological dose offsite and to personnel in the control room resulting from loss of coolant, main steam line break, steam generator tube rupture, loss of load, RCCA ejection and fuel handling accidents under Contract C-7693, Release Number 99-15. Additional DITs may be issued to provide other input required to perform the dose analyses.
DIT-B-00069-06 supercedes DIT-B-00069-05 in its entirety.

Source of Information: Engineering Judgement Used? ☒ Yes ☐ No
See attached table of input parameters and list of references.
Engineering judgement is used as a source of information for items C13 and L26 on the attached table of input parameters.

Distribution: Copy to Requestor Jill Monahan, Westinghouse
Copy to DIT Administrator File
Copy to Alan Meinershagen, AEP Solutions Teams
Copy to Ken Gerling, AEP Licensing
Original to NRM (Transmitted by DIT Administrator)

DESIGN INPUT LIST FOR D.C. COOK
OFFSITE & CONTROL ROOM DOSE ANALYSIS

COMMON TO ALL ANALYSES					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
C1	Core Power, MWt	3588.	Uprated core power per WCAP-14488, Table 2.1-1. The value is conservative compared to currently licensed values of 3250 and 3411 per U1/U2 Facility Operating Licenses. Values do not include 2% calorimetric error or pump heat.	Approved for Use	1 13
C2	Control room breathing rate, m^3/sec	3.47E-4	Murphy-Campe, 13th AEC Air Cleaning Conference	Approved for Use	15
C3	Off-site Breathing Rates (m^3/sec) 0-8 Hours 8-24 Hours 1-30 Days	 3.47E-4 1.75E-4 2.32E-4	Per RG 1.4, These values were developed from the average daily breathing rate [2 x 10(7) cm(3)/day] assumed in the report of ICRP, Committee II-1959.	Approved for Use	8
C4	CR Occupancy Factor 0-24 Hours 1-4 Days 4-30 Days	 1.0 0.6 0.4	Murphy-Campe, 13th AEC Air Cleaning Conference.	Approved for Use	15
C5	Control Room TEDE Dose Acceptance Criteria (Rem)	5.0	5 rem TEDE for Control Room dose will be considered the limit per the guidance being developed for the NUREG-1465 effort.	Approved for Use	9
C6	Atmospheric Dispersion Factors for Releases from Containment Building Surface to the, (sec/m^3) Site Boundary 0-2 Hours Low Population Zone 0-24 Hours 1-5 Days 5-30 Days	 3.15E-4 7.5E-5 2.6E-6 7.9E-7	UFSAR U2 Table 14.3.5-5	Unverified	
C7	Dose Conversion Factors for Nuclides, rem/Ci	See attached Table 1	CEDE and WB conversion factors from ICRP-30. Table 1 values are from Federal Guidance Report No 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," US EPA, 1988. Multiply by 3.7 x 10^{12} to convert to rem/Ci.	Approved for Use	18
C8	Control Room Emergency Volume Minimum Net Volume (ft^3) Maximum Gross Volume (ft^3)	 50,616 89,890	The control room emergency volumes are determined in calculation MD-12-HV-005-N. They include the control room, equipment room and computer room. Net volume is gross volume less cabinets and equipment.	Approved for Use	19

DESIGN INPUT LIST FOR D.C. COOK
OFFSITE & CONTROL ROOM DOSE ANALYSIS

COMMON TO ALL ANALYSES					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
C9	Normal Ventilation (SCFM) CR Filtered Makeup Air Flow Rate CR Filtered Recirc. Air Flow Rate Maximum CR Unfiltered Makeup Air Flow Rate CR Unfiltered Recirc. Air Flow Rate	 0.0 0.0 1000. 13400 ±10%	Calculation MD-12-HV-006-N establishes a minimum makeup air flow rate to maintain 0.0625 inches W.G. The value used, 1000 cfm, is greater than the minimum makeup air flow rate. 13,400 cfm from DCCHV12CR07N, page 13, restricted use, but not used in dose analysis. ±10% from ES-HVAC-0800QCN, Paragraph 4.11.4	Approved for use	20 45 52
C10	Emergency Ventilation (SCFM) Maximum CR Filtered Makeup Air Flow Rate Minimum CR Filtered Recirc. Air Flow Rate Maximum CR Unfiltered inleakage CR Unfiltered Recirc. Air Flow Rate	 1000/2000 4400/8800 98 13400 ±10%	Two entries are for single/dual pressurization fans running. Calculation MD-12-HV-006-N establishes a minimum makeup air flow rate to maintain 0.0625 inches W.G. The value used, 1000/2000 cfm, is greater than the minimum makeup air flow rate. The recirculation air flow rate is 5400 cfm less the makeup air flow rate per calculation MD-12-HV-006-N, item 4.3.4. Unfiltered inleakage determined by Tracer Gas Testing, Ref. 6. The limiting unfiltered inleakage is Unit 2 at 49±49 scfm. Unit 1 unfiltered inleakage of 144+24 scfm is not limiting after damper repair which eliminates 107-28 scfm for a net unfiltered inleakage after repair of 168-79, or 89 scfm in Unit 1. 13,400 cfm from DCCHV12CR07N, page 13, restricted use, but not used in dose analysis. ±10% from ES-HVAC-0800QCN, Paragraph 4.11.4	Approved for Use	6 20 45 52
C11	Control Room Iodine Removal Eff. with 1 Pressurization Fan Running (%) Elemental Methyl (organic) Particulate	 95 95 98	Tech Spec 3/4.7.5.1 change required for charcoal and particulate filter efficiency testing. 98% particulate efficiency is based on a DOP test penetration of less than 1% at rated flow with a safety factor of 2. See CR 99-2037 for particulate filtration concern. Action item in CR 99-12640 to change TS.	Approved for Use	13 14 61
C12	Control Room Iodine Removal Eff. with 2 Pressurization Fans Running (%) Elemental Methyl (organic) Particulate	 80 80 98	Calculation MD-12-HV-0009-N unverified due to TS 3/4.7.5.1 not referencing ASTM D 3803-89 (re-approved 1995) and tracked by CR 99-19061. Action item in CR 99-12640 to change TS. Particulate efficiency is not affected by flow rate.	Approved for Use	29 50

DESIGN INPUT LIST FOR D.C. COOK
OFFSITE & CONTROL ROOM DOSE ANALYSIS

COMMON TO ALL ANALYSES					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
C13	Delay to Switch CR HVAC from Normal Operation to Emergency Operation after receiving an isolation signal (sec)	66	This is the time it will take from the initiating S-signal until the emergency operation of the HVAC system is in operation. Dampers are being purchased under DCP-574 with an opening/closing time of less than 30 seconds. UFSAR Table 7.2-6 lists instrument response times. All are less than 6 seconds. UFSAR Section 14.1 uses 30 seconds for EDG startup. Engineering judgement is used to apply a time, similar to the times supported by analysis and reported in the UFSAR, to the CR HVAC system.	Approved for Use	7 51
C14	Time to turn off one pressurization fan (minutes)	0/30/120	Per conversation with R. Foster, July 12, 1999, 30 minutes to turn off one fan normally, 120 minutes considering failure to manually turn off one fan. Having one fan off initially may be conservative in some cases. Also consider failure of second fan to continue to run.	Approved for Use	63

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LARGE-BREAK LOCA					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
L1	Core Release Fraction Iodine Noble Gases Other Nuclides	Consistent with NUREG-1465	Time dependent release will be modeled as defined in NUREG-1465	Approved for Use	21
L2	Iodine Plate-out Fraction on Containment Surfaces	N/A	No instantaneous plate-out on surfaces is credited in NUREG-1465.	Approved for Use	21
L3	Initial Iodine Species in Containment (%) Elemental Methyl (organic) Particulate	 4.85 0.15 95.0	Species distribution identified in NUREG-1465 and interpreted in working draft of Regulatory Guide 1081, Appendix A.	Approved for Use	21 22
L4	Containment Leak Rate (weight %/day) 0-280 Hours > 280 Hours	 0.25 0.125	0.25 wt%/day leak rate per Tech Spec 3/4.6.1.2. Containment pressure 25% of limit 280 hours into event per SECL 99-076 Figure 3.4-1 to support 50% reduction in containment leak rate. More conservative than 50% reduction at 24 hours per RG 1.4. CR-99-12890 identifies condition.	Approved for Use	8 13 53 54
L5	Containment Net Volumes (ft^3) upper containment ice condenser (w/o ice) lower containment, active lower containment, dead end	1,255,074 734,320 162,060 296,767 61,927	Best estimate values from DIT-B-00010-01 reference SD-990618-003, rev 0. Values should be reduced by 1% as specified by the DIT for minimum values. The ice condenser volume, below the intermediate deck doors, net of equipment w/o ice is 162,060 per letters AEP-99-432 and AEP-99-401 (73% of 222,000).	Approved for Use	23 67 68
L6	Deleted				
L7	Containment spray (CS) injection flow rate, gpm upper (E/W) lower (E/W) annulus (E/W)	 2942 1960/1977 728/706 264/259	These spray flow rates do not consider obstruction of spray patterns. Values are from TH-98-09, rev 2. Obstructions are accounted for by taking a 10% reduction in spray removal coefficients per AEP-99-487 (CR 99-5939).	Approved for Use	24 46
L8	Containment spray recirc. flow rate, gpm upper (E/W) lower (E/W) annulus (E/W)	2942 1960/1977 728/706 264/259	These spray flow rates do not consider obstruction of spray patterns. Values are from TH-98-09, rev 2. Obstructions are accounted for by taking a 10% reduction in spray removal coefficients in AEP-99-487 (CR 99-5939).	Approved for Use	24 46
L9	RHR spray flowrate (gpm)	1850	Westinghouse letter report AEP-99-487 references Westinghouse letter SAE/FSE-AEP/AMP-0287, 12/2/97 for RHR spray flow rate.	Approved for Use	46

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LARGE-BREAK LOCA					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
L10	Spray fall height (ft) upper lower annular RHR	 59.7 26.8 4.1 41.6	Issues exist on the containment spray fall heights as part of the CTS ESRR. 1973 calculation as referenced in CR 99-7682. CR 99-21135 identifies a concern with header pressure. Fall heights calculated in SD-990809-001-N, rev. 1 address both CRs.	Approved for Use	25
L11	CS injection flow initiated, sec	115	Includes 30 sec diesel generator start time Use 115 s for initiation of all containment spray. 115s per UFSAR Table 14.3.5-5 AEP/W SGTP-16, Table 3.5-1, item 7b, 115s for lower CS, 104s for upper CS based on Calc. N940201	Approved for Use	4
L12	RHR spray flow initiated, sec	4500	1.25 hours (4500 seconds) is assumed for initiation of RHR sprays. Westinghouse letter report AEP-99-487 references the Westinghouse Containment Integrity analysis documented in WCAP-15302 for the time at which RHR sprays are initiated.	Approved for Use	46
L13	CS injection flow duration, min.	75	1.25 hours (75 minutes) based on RWST drain down time assuming a single train of ECCS in operation and minimum suction flow rates per Westinghouse letter SAE/FSE-AEP/AMP-0786.	Approved for Use	46
L14	CS recirculation and RHR flow termination, hours from event initiation	6	Westinghouse letter report AEP-99-487.	Approved for Use	46
L15	Time delay between CTS injection and recirculation flows (min)	5	No change per memo from Jim Abshire.	Approved for Use	37
L16	Air stream flow rates, cfm lower to upper, 0-150 sec. lower to upper through ice, >150 sec between active and dead end lower	 0 39,000 100 to 900	Lower to upper, 0-150 seconds value is limiting, therefore assume zero. CEQ starts on hi pressure signal, 1.1 psig, within 150 seconds. 150 seconds verified per DIT-B-00003-06. 39,000 cfm flow rate from drawing 5147A. Sensitivity analysis in AEP-99-487 uses 100 to 900 cfm for exchange from lower to annular region based on hydrogen skimmer fan (HV-CEQ-1(2)) flow rate from drawings 5147 and 5147A. Not clear if low or high is conservative. The Westinghouse report concludes that an intermediate value of 500 cfm is conservative.	Approved for Use	7 17 31 32 46 47
L17	Upper containment CS iodine removal coefficients, hr^{-1} elemental during injection elemental during recirculation particulate with/without RHR	 20.0 8.9 7.3/4.3	Westinghouse letter report AEP-99-487, Table 12, documents spray removal coefficients calculated using methods described in the report. Elemental during injection limited to 20 hr^{-1} by SRP Section 6.5.2 Rev. 2. Particulate coefficient reduced to 10% of calculated value when particulate DF reaches 50 hr^{-1}	Approved for Use	46

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LARGE-BREAK LOCA					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
L18	Lower containment CS iodine removal coefficients, hr^{-1} elemental during injection elemental during recirculation particulate	 12.0 4.5 2.3	Westinghouse letter report AEP-99-487, Table 12, documents spray removal coefficients calculated using methods described in the report. Elemental during injection limited to 20 hr^{-1} by SRP Section 6.5.2 Rev. 2. Particulate coefficient reduced to 10% of calculated value when particulate DF reaches 50 hr^{-1}	Approved for Use	46
L19	CS elemental iodine DF upper lower	 ≤200 ≤200	Westinghouse letter report AEP-99-487 documents that the decontamination factor of 200 is reached 2.6 hours into the event. Limited to 200 hr^{-1} based on SRP Section 6.5.2 Rev. 2	Approved for Use	46
L20	Nominal containment temperature, °F upper lower	 100 120	Tech Spec 3/4.6.1.5 maximum average air temperature values	Approved for Use	13
L21	Maximum containment temperature, °F	Not Used in Analysis	The Westinghouse radiological dose anlaysis method does not use containment temperature. Spray flow rates are volumetric and not dependent on spray temperature.	N/A	N/A
L22	Injection Spray temperature, °F upper lower	 105 105	Procedures 1(2) OHP 4030.STP.030 establish maximum indicated RWST temperature of 93.5°F and procedures ECP 1-I9-01 and EC 2-I9-02 establish maximum instrument uncertainty of 4.65°F. An additional 6.85°F margin is added for conservatism. Plant data collected in 1992 using procedures 1-OHP 4030.STP.030 and 2-OHP 4030.STP.030, "Shift Surveillance Checks" show max temp is 99.2°F. TS 3.1.2.8.b.3 is being changed to establish 100°F as the maximum RWST temperature. 105°F is conservative.	Approved for Use	4 40 41 56 57 58 59 64
L23	Ice condenser iodine removal efficiency	0.3	SRP 6.5.4 Rev. 3. Efficiency is modeled after CEQ fans start (see note in item L16) and until first ice bed meltout or until element iodine DF = 200 per SRP 6.5.2.	Approved for Use	14

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LARGE-BREAK LOCA

No.	Design and Licensing Parameters	AEP Prop. Value	Notes	Status	Ref.
L24	Containment sump water volume, long term steady state, gal	514,642	A minimum volume is conservative because it results in the highest radionuclide concentrations in the sump. Assume radionuclides are homogeneously mixed in the active and inactive sump water. RWST 314,000 per DIT-B-00003-06, item 6d Accumulators 27,556 per AEP-99-266 Ice Condenser 255,800 per SECL-99-076 and FAI calc RCS 35,081 per TH-97-12 Subtract the volume of the reactor cavity (117,795 gal. per DIT-B-00033-00) because it may be at a lower radionuclide concentration and may not be well mixed with the rest of the sump. 314,000 gal. is a design requirement tracked by CR 99-21216. 27,556 gal. is 4 x 921 ft^3 per accumulator x 7.481 gal/ft^3. 255,800 gal. is (2.2 Mlbm – 65,000 lbm) x 7.481 gal/ft^3 x .016019 ft^3/lbm. 2.2 Mlbm of ice is based on SECL-99-076, paragraph 3.4.6. It is the minimum ice mass from the peak containment pressure analysis and bounds minimum ice mass of 2.6 Mlbm in TS 3.6.5.1. 65,000 lbm bounds the mass of ice remaining after 2 weeks per FAI calc as described in item L40. The specific volume, .016019 ft^3/lbm, is conservatively taken as the minimum possible for water. 35,081 gal. is a conservative RCS volume from restricted calculation TH-97-12. It is less than the reactor vessel volume alone (4748 ft^3 x 7.481 gal/ft^3) from AEP-99-370.	Approved for Use	3 13 47 48 53 54 60 65 66
L25	Containment sump water volume at time of switchover to sump recirculation, gal	216,000	Sump volume associated with 602' 10" (see SECL-99-076), the minimum volume to initiate switchover to sump recirculation per Calculation No. DC-D-3200S-227 in DIT-B-00033-00. The RWST is not completely injected and not all of the ice is melted. The remainder of these sources must be added as a function of time after switchover to recirculation.	Approved for Use	48 49 53 54
L26	Containment sump solution pH	7.0	DIT-B-00236-04 documents that sump pH will be ≥7.0 during recirculation based on engineering judgement supported by S&L calculation MD-12-CTS-118-N transmitted to AEP for owner's acceptance.	Approved for Use	10
L27	ECCS leakage source term	40% core iodine	Time dependent release will be modeled based on NUREG-1465, Table 3.13 gap release plus early in-vessel release for halogens.	Approved for Use	21

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LARGE-BREAK LOCA

No.	Design and Licensing Parameters	AEP Prop. Value	Notes	Status	Ref.
L28	ECCS leakage rate (gpm) ELO valve lift for 8 hrs Effective unfiltered via Unit vent for duration	 0 0.2	ELO valve lift leak rate is 0 gpm for 8 hours from MD-12-ECCS-007-N. Westinghouse letter report AEP-99-487 supports a 0.2 gpm ECCS leak rate. Acceptable results are achieved and the plant is expected to be capable of maintaining leakage below this limit. Report AEP-99-487 is owner accepted.0.2 gpm is an assumed leak rate conservatively assumed to be released unfiltered from the Unit vent. Sensitivity runs are required to determine the doses for a filtered release and for a release from the RWST. CR 99-3135 and CR 98-3076	Approved for Use	55
L29	Time of recirculation initiation, min containment leakage case ECCS leakage case	 80 0	Westinghouse letter report AEP-99-487. For the containment leakage case switchover to sump recirculation is initated at 1.25 hours with a 5 minute delay to remove spray. For the ECCS leakage case it is conservative to assume recirculation starts immediately.	Approved for Use	5 46
L30	ECCS passive failure leak rate beginning at 24 hours after LOCA, gpm for 30 min	50	Per SRP 15.6.5, App. B, Rev. 1, and DG-1081, App. A, item 5c, the passive failure case is only considered for plants that do not have an ESF atmosphere filtration system. CR 99-16309 questions if the ESF vent system covers all areas.	Approved for Use	14 22
L31	ECCS leakage/passive failure iodine airborne fraction	0.1, 0.01, 0.001, and 0.0001	Use a range of values between 0.1 per SRP 15.6.5, App. B, Rev. 1 and working draft reg guide DG-1081, Appendix A and 0.0001 per AERER4887 and UFSAR 14.3.5.	Approved for Use	14 22 44
L32	Aux. building filtration for ECCS leakage Elemental Methyl (organic)	 90% 90%	Assume all ECCS leakage is unfiltered. TS 3/4.7.6.1	Approved for Use	13
L33	Purge system flowrate, cfm	16,000	Purge system flow rate is 16,000 cfm per fan from drawing 5147A. Only 1 valve is open at a time per TS 3.6.1.7. CR 99-3296	Approved for Use	13 31 32
L34	Delay for purge system isolation (sec)	5.0	For containment purge case; Specification DCCNEMP306QCN, Rev. 0, "Containment Boundaries and Test Connections", 10-31-95. CR 99-3296	Approved for Use	27
L35	Atmospheric Dispersion Factors for Release from Unit Vent to Control Room Intake (X/Q), sec/m^3 0-2 hr 2-8 hr 8-24 hr 1-4 days 4-5 days 5-30 days	 1.74E-3 1.00E-3 4.19E-4 3.09E-4 2.69E-4 2.69E-4	RD-99-05 is EQE/PLG Calculation C-254055-05.	Approved for Use	28

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LARGE-BREAK LOCA

No.	Design and Licensing Parameters	AEP Prop. Value	Notes	Status	Ref.
L36	Offsite TEDE dose acceptance criteria (rem)	25.0	NRC proposed rulemaking as documented in SECY-98-289.	Approved for Use	9
L37	Deleted				
L38	Duration of CR & LPZ dose accumulation	30 days	Murphy-Campe, 13th AEC Air Cleaning Conference	Approved for Use	15
L39	Iodine Species in ECCS release (%) Elemental Methyl (organic)	 100 0	Not particulate. Distribution between elemental or organic has no impact.	Approved for Use	22
L40	Mass of ice remaining (Mlbm) 0 hours 2 hours 8 hours 24 hours 48 hours 100 hours 150 hours 200 hours 250 hours 300 hours 2 weeks	 2.2 1.6 1.4 1.2 1.0 0.75 0.5 0.35 0.25 0.15 0.065	Calculation FAI/99-86 done with one train of safeguards. The conservative direction is to maximize the mass of ice remaining because it minimizes the sump volume and maximizes the radionuclide concentration in the sump water. The proposed values are conservative compared to FAI/99-86 including differences in initial ice mass as shown in the following table; Time (hrs) / FAI/99-86 / Adj. Value / Prop. Value / Margin 0 / 2.11 / 2.20 / 2.20 / 0.00 2 / 1.42 / 1.51 / 1.60 / 0.09 8 / 1.15 / 1.24 / 1.40 / 0.16 24 / 0.78 / 0.87 / 1.20 / 0.33 48 / 0.44 / 0.53 / 1.00 / 0.47 100 / 0.15 / 0.24 / 0.75 / 0.51 etc.	Approved for Use	60
L41	Atmospheric Dispersion Factors for Release from Containment Building Surface to Control Room Intake, (sec/m^3) 0-2 Hours 2 to 8 Hours 8 to 24 Hours 1 to 4 Days 4 to 30 Days	 8.95E-4 6.91E-4 2.79E-4 2.23E-4 2.63E-4	RD-99-05 is EQE/PLG Calculation C-254055-05.	Approved for Use	28
L42	Atmospheric Dispersion Factors for Release from RWST to Control Room Intake (X/Q), sec/m^3 0-2 hr 2-8 hr 8-24 hr 1-4 days 4-5 days 5-30 days	 1.16E-3 7.92E-4 3.68E-4 2.52E-4 1.97E-4 1.97E-4	RD-99-05 is EQE/PLG Calculation C-254055-05. The RWST is the release point for that portion of the ECCS leakage that returns to the RWST. AEP requires a calculation to quantify the difference in control room dose between a release from the unit vent and a release from the RWST.	Approved for Use	28

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LARGE-BREAK LOCA

No.	Design and Licensing Parameters	AEP Prop. Value	Notes	Status	Ref.
L43	CTS spray coverage (% Vol. Unsprayed)		Per calculation SD-990809-006-N and CR 99-5921	Approved for Use	25
	upper	30.5	0.305 = 179634/(409608+179634)		
	lower	30.4	0.304 = 47795/(47795+109369)		
	annular	75.5	0.755 = 19140/(19140+6199)		
L44	RHR spray coverage (% Vol. Unsprayed)	53.7	0.537 = 317044/(317044+273767)	Approved for Use	25

SMALL-BREAK LOCA

Per DG-1081 Appendix A, gap fractions from Table 3 can be used for SBLOCA if no fuel melt is projected. LOCA PCT analysis demonstrates that no fuel melt is predicted for any LOCA events. Containment sprays are conservatively assumed to not actuate. All other protective action is the same as large-break LOCA.

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Main Steam Line Break					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
M1	Primary Coolant Activity (μCi/gram I-131 equivalent) Normal Iodine TS Limit Pre-accident Iodine spike	 1.0 60.0	Tech Spec 3.4.8 and Figure 3.4-1	Approved for Use	13
M2	Primary Coolant Noble Gas Activity Prior to Accident	Equivalent to 1% defects	Approximately equal to Tech Spec 3.4.8 of 100/E-bar.	Approved for Use	13
M3	Secondary Coolant Activity (μCi/gram I-131 equivalent)	0.1	Tech Spec 3.7.1.4	Approved for Use	13
M4	Number of Rods which enter DNB and have Fuel Centerline Melt (%)	0	No fuel damage is calculated for SLB. This is a Westinghouse safety analysis acceptance criteria that is confirmed in an internal Westinghouse calculation.	Approved for Use	46
M5	Primary to Secondary Leakage Faulted Intacts	 500 gpd - Unit 2 150 gpd - Unit 1 940 gpd - Unit 2 450 gpd - Unit 1	For Unit 2, values are taken from Tech Spec 3.4.6.2. The intact steam generators have a total leak rate of 1 gpm - 500 gpd through faulted, or 940 gpd. For Unit 1, values are from 3.4.6.2. The intact steam generators have a total leak rate of 600 gpd - 150 gpd through faulted, or 450 gpd. . CR 99-16385	Approved for Use	13
M6	SG Iodine Partition Factor Intact SG Faulted SG	 0.01 1.0	The values are consistent with SRP Section 15.1.5, Appendix A. The faulted SG value reflects dryout of the SG shortly after accident initiation. Also in DG-1081.	Approved to Use	14
M7	RCS Letdown Flow Flow Rate (gpm) Temperature (°F) DF	 120 127 infinite	This is the maximum purification flow rate of two orifices (75+45) and nominal temperature per Westinghouse System Description, Ref 33 Page 79 and AEP System Description, Ref 35 Pages 3, 8 and 9. RCS Letdown Flow is used together with RCS leakage to determine the iodine appearance rate. An infinite DF is used to ensure a conservative value. Letdown is considered isolated following the accident.	Approved for Use	33 34 35
M8	Primary coolant losses from leakage (gpm)	12	Based on leakage at the maximum TS 3.4.6.2 limits of 1.0 gpm unidentified leakage, 10 gpm of identified leakage, and 1 gpm of P/S leakage. This leakage, combined with the letdown flow cleanup, is used to determine the iodine appearance rate. Leakage is based on cold conditions.	Approved for Use	13

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Main Steam Line Break					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
M9	Steam Released to Environment, lbm Faulted SG Intact SG 0-2 hours 2-8 hours 8-24 hours 1-7 days 7-30 days	 161,000 456,000 1,186,000 1,347,000 8,844,000 33,519,000	Westinghouse letter report AEP-99-487.	Approved for Use	46
M10	Time to generate signal to switch from CR HVAC Mode 1 to Mode 2 operation (sec)	300	Westinghouse letter report AEP-99-487. An S-signal occurs within the first minute of the event and the emergency HVAC system is conservatively assumed to start 5 minutes after event initiation.	Approved for Use	46
M11	Atmospheric Dispersion Factors for Release from PORVs to Control Room Intake (sec/m^3) 0-2 hr 2-8 hr 8-24 hr 1-4 days 4-5 days 5-30 day	 4.55E-4 2.87E-4 1.23E-4 9.60E-5 6.83E-5 6.83E-5	RD-99-05 is EQE/PLG Calculation C-254055-05. The PORV release point should be used for all intact SG releases. Increase values listed here by 10% to account for partial entrainment in building wake. U2 West PORV to U1 Intake U2 West PORV to U1 Intake U1 East PORV to U1 Intake U1 East PORV to U1 Intake U2 West PORV to U1 Intake U2 West PORV to U1 Intake	Approved for Use	28
M12	Offsite TEDE Dose Acceptance Criteria (Rem) Pre-accident Iodine Spike Case Accident initiated Iodine Spike Case	 25. 2.5	100% of LOCA limit for pre-accident iodine spike and 10% of LOCA limit for accident initiated spike case. NRC proposed rulemaking as documented in SECY-98-289. Per RG 15.1.5, Section II, a small fraction of 10 CFR 100 (i.e. 10%). the same acceptance criteria are expected to be applied in DG-1081	Approved for Use	9 14
M13	Atmospheric Dispersion Factors for Release from Containment Building Surface to Control Room Intake, (sec/m^3) 0-2 Hours 2 to 8 Hours 8 to 24 Hours 1 to 4 Days 4 to 30 Days	 8.95E-4 6.91E-4 2.79E-4 2.23E-4 2.63E-4	RD-99-05 is EQE/PLG Calculation C-254055-05	Approved for Use	28

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Main Steam Line Break					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
M1 4	Atmospheric Dispersion Factors for Release from Unit Vent to Control Room Intake (X/Q), sec/m^3		RD-99-05 is EQE/PLG Calculation C-254055-05. The Unit vent is the release point for all auxiliary building ventilation systems. Unit vent release should be used for the faulted SG release because Chi/Q values are the maximum , most conservative, of the PORV, Unit vent and containment building release points.	Approved for Use	28
	0-2 hr	1.74E-3			
	2-8 hr	1.00E-3			
	8-24 hr	4.19E-4			
	1-4 days	3.09E-4			
	4-5 days	2.69E-4			
	5-30 days	2.69E-4			

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SGTR Radiological Input					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
G1	Primary Coolant Activity (µCi/gram dose equivalent I-131) Normal Iodine TS Limit Pre-accident Iodine spike TS Limit	 1.0 60.0	 Tech Spec 3.4.8 and Figure 3.4-1	Approved for Use	13
G2	Primary Coolant Noble Gas Activity Prior to Accident	Equivalent to 1% defects	Approximately equal to Tech Spec 3.4.8 of 100/E-bar.	Approved for Use	13
G3	Secondary Coolant Activity (µCi/cc I-131 equivalent)	0.1	Tech Spec 3.7.1.4	Approved for Use	
G4	Primary to Secondary Leakage Limit Unit 1 (gpd) Unit 2 (gpm)	 600 1	 This value is taken from Unit 1 Tech Spec 3.4.6.2 This value is taken from Unit 2 Tech Spec 3.4.6.2	Approved for Use	13
G5	SG Iodine Partition Factor	0.01	SRP Section 15.6.3 Revision 2. DG-1081, Appendix F confirms that an iodine partitioning coefficient of 100 is appropriate.	Approved for Use	14 22
G6	RCS Letdown Flow Flow Rate (gpm) Temperature (°F) DF	 120 127 infinite	This is the maximum purification flow rate of two orifices (75+45) and nominal temperature per Westinghouse System Description, Ref 33 Page 79 and AEP System Description, Ref 35 Pages 3, 8 and 9. RCS Letdown Flow is used together with RCS leakage to determine the iodine appearance rate. An infinite DF is used to ensure a conservative value. Letdown is considered isolated following the accident.	Approved for Use	33 34 35
G7	Primary coolant losses from leakage (gpm)	12	Based on leakage at the maximum TS 3.4.6.2 limits of 1.0 gpm unidentified leakage, 10 gpm of identified leakage, and 1 gpm of P/S leakage. This leakage, combined with the letdown flow cleanup, is used to determine the iodine appearance rate. Leakage is based on cold conditions.	Approved for Use	13
G8	Rupture Flow (lbm) Break Flow Flashing Fraction (%) Steam Released to Environment (lbm) Ruptured SG, 0-2 hours Intact SG 0-2 hours 2-8 hours 8-24 hours 1-7 days 7-30 days	162,000 16.77 73,000 565,000 1,505,000 1,482,000 9,729,000 36,871,000	Westinghouse letter report AEP-99-487.	Approved for Use	46

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SGTR Radiological Input					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
G9	Time to generate signal to switch from CR HVAC Mode 1 to Mode 2 operation (sec)	420	Westinghouse letter report AEP-99-487. An S-signal occures within the first 5.6 minutes of the event and the emergency HVAC system is conservatively assumed to start 7 minutes after event initiation.	Approved for Use	46
G10	Atmospheric Dispersion Factors for Release from PORVs to Control Room Intake (sec/m^3)		RD-99-05 is EQE/PLG Calculation C-254055-05. Increase values listed here by 10% to account for partial entrainment in building wake.	Approved for Use	28
	0-2 hr	4.55E-4	U2 West PORV to U1 Intake		
	2-8 hr	2.87E-4	U2 West PORV to U1 Intake		
	8-24 hr	1.23E-4	U1 East PORV to U1 Intake		
	1-4 days	9.60E-5	U1 East PORV to U1 Intake		
	4-5 days	6.83E-5	U2 West PORV to U1 Intake		
	5-30 day	6.83E-5	U2 West PORV to U1 Intake		
G11	Offsite TEDE Dose Acceptance Criteria (Rem)		100% of LOCA limit for pre-accident iodine spike and 10% of LOCA limit for accident initiated spike case.	Approved for Use	9
	Pre-accident Iodine Spike Case	25.			
	Accident initiated Iodine Spike Case	2.5			

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SGTR Thermal and Hydraulics Input*					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
*Analyses will be performed for both Unit 1 and Unit 2. The limiting break flow from all cases and the limiting steam release to the atmosphere from all cases (these will be different cases) will be used in the calculation of the offsite site and control room doses.					
H1	Core Power (MWt)	3588.	Uprated core power per WCAP-14488, Table 2.1-1. The value is conservative compared to currently licensed values of 3250 and 3411 per U1/U2 Facility Operating Licenses. Values do not include 2% calorimetric error or pump heat.	Approved for Use	1 13
H2	Reactor Coolant System Pressure (psig)	U1 ≥2050 U2 ≥2200	TS 3.2.5	Approved for Use	13
H3	Vessel Average Temperature (°F) Unit 1 Unit 2	 547 to 576.3 547 to 581.3	High and Low Tavg ranges will be evaluated Unit 1 at 3262 MWt NSSS Power per WCAP-14286 Unit 2 at 3600 MWt NSSS Power per WCAP-12135 and Vantage 5 RTSR	Approved for Use	2 38 39
H4	No Load Temperature (°F)	547	Per WCAP-14286 for Unit 1 and WCAP-14488 for Unit 2	Approved for Use	1 2
H5	Steam Pressure (psia)	888	Westinghouse letter report AEP-99-487. The steam conditions are selected to result in the highest post-trip flashing fraction.	Approved for Use	46
H6	Steam Temperature (°F)	615.2	Westinghouse letter report AEP-99-487. The steam conditions are selected to result in the highest post-trip flashing fraction.	Approved for Use	46
H7	Steam Gen. Tube Plugging (avg%/peak%) Unit 1 Unit 2	 30/30 10/15	 For Unit 1 at 3262 MWt NSSS Power per WCAP-14286 For Unit 2 3600 MWt NSSS Power per WCAP-12135 and Vantage 5 RTSR	Approved for Use	2 38 39
H8	Safety Injection Setpoint - Pzr Pressure (psig) Unit 1 Unit 2	 1815±13.7% 1900±13.7%	 Per WCAP-13055, Table 3-19, item 18 Per WCAP-13801, Table 3-19, item 18	Approved for Use	42 43
H9	Lowest Steam Generator Safety Valve Setpoint (psia)	1080	Tech Spec Value from Table 4.7-1	Approved for Use	13
H10	SG Safety Valve Blowdown (%)	15	Maximum blowdown value for Dresser model safety valves per WCAP 12539.	Approved for Use	36
H11	SG Safety Valve Tolerance (%)	3	3% tolerance will be added to 15% blowdown to provide conservative results. See Tech Spec Table 4.7-1	Approved for Use	13

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SGTR Thermal and Hydraulics Input*

No.	Design and Licensing Parameters	AEP Prop. Value	Notes	Status	Ref.
H12	Safety Injection Flow Rates (lbm/sec as a function of RCS pressure in psig) 1700 1800 1900 2000 2100 2200 2300	 81.82 78.35 74.71 70.87 66.80 62.42 57.67	Westinghouse letter FRSS/SS-AEP-5253 or WCAP-14286, Table 4.4-19 maximum flow with no spilling.	Approved for Use	2 46
H13	RHR Cut-in Pressure (psia)	Atmospheric	The SGTR analysis will assume that conditions in the primary side (for the long term steam releases) reach 212 deg F per DG-1081, Appendix F, paragraph 5.c.	Approved for Use	22
H14	RHR Cut-in Temperature (°F)	212	See above.	Approved for Use	22
H15	RHR Cut-in Time (Days)	30	Westinghouse letter report AEP-99-487. The SGTR analysis assumes that the steam releases from the secondary side continue for the 30 day duration of the event.	Approved for Use	46
H16	Minimum Auxiliary Feedwater Flow (gpm)	800	Per TS Bases 3/4.7.1.2, 450 gpm at 1065 psig. Use 400 gpm per pump for conservatism. Assumes 2 motor driven auxiliary feedwater pump operating. Consistent with the assumption that auxiliary feedwater is initiated off loss of offsite power.	Approved for Use	13

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FUEL HANDLING ACCIDENT					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
F1	Power Peaking Factor	1.65	Consistent with Reg. Guide 1.25. Bounds 1.62 value used in LOCA PCT analysis per WCAP-14488, Table 3.2-2.	Approved for Use	1 30
F2	Decay Time (Hours)	168	Bounds Tech Spec 3.9.3 for Refueling time of 168 hours	Approved for Use	13
F3	Gap Activity (%) Iodine 131 All other iodine Kr-85 All other Noble Gases Alkali Metals	 12 10 15 10 10	Per NRC direction on non-LOCA events Table 3	Approved for Use	22
F4	Number of Assemblies Damaged	1	The design basis accident is the complete rupture of the highest rated spent fuel assembly per NRC SER N95091.	Approved for Use	62
F5	Iodine Chemical Species (%) Elemental Methyl	 99.75 0.25	Per Reg Guide 1.25 and working draft Reg Guide DG-1081.	Approved for Use	30 22
F6	Pool Scrubbing Factor for Iodine Elemental Methyl	 500 1	Per draft Reg Guide DG-1081. Taken together with the above identified species split for iodine, these values correspond to an "effective DF of 200".	Approved for Use	22
F7	Release duration (Hours)	2	Per working draft Reg Guide DG-1081.	Approved for Use	22
F8	Offsite TEDE Dose Acceptance Criteria (Rem)	6.25	25% of 25 rem LOCA limit, definition of well within per SRP 15.7.4	Approved for Use	9 14
F9	SFP Sweep System Filter Efficiency (%) Elemental Particulate Methyl	 N/A N/A N/A	No credit for spent fuel pool ventilation will be assumed.	Approved for Use	n/a

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FUEL HANDLING ACCIDENT					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
F10	Atmospheric Dispersion Factors for Release from Containment Building Surface to Control Room Intake,, (sec/m^3) 0-2 Hours 2 to 8 Hours 8 to 24 Hours 1 to 4 Days 4 to 30 Days	 8.95E-4 6.91E-4 2.79E-4 2.23E-4 2.63E-4	RD-99-05 is EQE/PLG Calculation C-254055-05.	Approved for Use	28
F11	Atmospheric Dispersion Factors for Release from Unit Vent to Control Room Intake (X/Q), sec/m^3 0-2 hr 2-8 hr 8-24 hr 1-4 days 4-5 days 5-30 days	 1.74E-3 1.00E-3 4.19E-4 3.09E-4 2.69E-4 2.69E-4	RD-99-05 is EQE/PLG Calculation C-254055-05. The unit vent is the release point for all auxiliary building ventilation systems.	Approved for Use	28

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Loss of Load					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
T1	Primary Coolant Activity (μCi/gram I-131 equivalent) Normal Iodine TS Limit Pre-accident Iodine spike	 1.0 60.0	 Tech Spec 3.4.8 and Figure 3.4-1	Approved for Use	13
T2	Primary Coolant Noble Gas Activity Prior to Accident	Equivalent to 1% defects	Approximately equal to Tech Spec 3.4.8 of 100/E-bar.	Approved for Use	13
T3	Secondary Coolant Activity (μCi/cc I-131 equivalent)	0.1	Tech Spec 3.7.1.4	Approved for Use	13
T4	Primary to Secondary Leakage Limit Unit 1 (gpd) Unit 2 (gpm)	 600 1	 This value is taken from Unit 1 Tech Spec 3.4.6.2 This value is taken from Unit 2 Tech Spec 3.4.6.2	Approved for Use	13
T5	SG Iodine Partition Factor for non-flashed break flow	0.01	The values are consistent with SRP Section 15.6.3. DG-1081, Appendix G confirms that an iodine partitioning coefficient of 100 is appropriate.	Approved for Use	14 22
T6	Steam Released to Environment (lbm) Intact SGs 0-2 hours 2-8 hours 8-24 hours 1-7 days 7-30 days	 738,000 1,345,000 1,590,000 11,030,000 36,871,000	Westinghouse letter report AEP-99-487	Approved for Use	46
T7	RCS Letdown Flow Flow Rate (gpm) Temperature (°F) DF	 120 127 infinite	This is the maximum purification flow rate of two orifices (75+45) and nominal temperature per Westinghouse System Description, Ref 33 Page 79 and AEP System Description, Ref 35 Pages 3, 8 and 9. RCS Letdown Flow is used together with RCS leakage to determine the iodine appearance rate. An infinite DF is used to ensure a conservative value. Letdown is considered isolated following the accident.	Approved for Use	33 34 35
T8	Primary coolant losses from leakage (gpm)	12	Based on leakage at the maximum TS 3.4.6.2 limits of 1.0 gpm unidentified leakage, 10 gpm of identified leakage, and 1 gpm of P/S leakage. This leakage, combined with the letdown flow cleanup, is used to determine the iodine appearance rate. Leakage is based on cold conditions.	Approved for Use	13
T9	Time to generate signal to switch from CR HVAC Mode 1 to Mode 2 operation (sec)	N/A	Initially no switchover to emergency HVAC will be assumed	n/a	n/a

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Loss of Load					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
T10	Atmospheric Dispersion Factors for Release from PORVs to Control Room Intake (sec/m^3)		RD-99-05 is EQE/PLG Calculation C-254055-05. Increase values listed here by 10% to account for partial entrainment in building wake.	Approved for Use	28
	0-2 hr	4.55E-4	U2 West PORV to U1 Intake		
	2-8 hr	2.87E-4	U2 West PORV to U1 Intake		
	8-24 hr	1.23E-4	U1 East PORV to U1 Intake		
	1-4 days	9.60E-5	U1 East PORV to U1 Intake		
	4-5 days	6.83E-5	U2 West PORV to U1 Intake		
	5-30 day	6.83E-5	U2 West PORV to U1 Intake		
T11	Offsite TEDE Dose Acceptance Criteria (Rem)		100% of LOCA limit for pre-accident iodine spike and 10% of LOCA limit for accident initiated spike case.	Approved for Use	9 14
	Pre-accident Iodine Spike Case	25.			
	Accident initiated Iodine Spike Case	2.5	SRP 15.6.3 acceptance criteria is a small fraction, or 10% of LOCA limit.		

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RCCA EJECTION					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
R1	Primary Coolant Activity (μCi/gram I-131 equivalent) Pre-accident Iodine spike	60.0	Tech Spec 3/4.4.8 Figure 3.4-1	Approved for Use	13
R2	Primary Coolant Noble Gas Activity Prior to Accident	Equivalent to 1% defects	Approximately equal to Tech Spec 3.4.8 of 100/E-bar.	Approved for Use	13
R3	Secondary Coolant Activity (μCi/cc I-131 dose equivalent)	0.1	Current TS 3.7.1.4 Limit.	Approved for Use	13
R4	Primary to Secondary Leakage Limit Unit 1 (gpd) Unit 2 (gpm)	 600 1	 This value is taken from Unit 1 Tech Spec 3.4.6.2 This value is taken from Unit 2 Tech Spec 3.4.6.2	Approved for Use	13
R5	SG Iodine Partition Factor	0.01	SRP Section 15.6.3, Revision 2. DG-1081, Appendix H confirms that an iodine partitioning coefficient of 100 is appropriate for the rod ejection accident.	Approved for Use	14 22
R6	Rods in DNB (% failed fuel)	15	Westinghouse letter report AEP-99-487. WCAP-7588, Rev. 1 specifies 10% but a value of 15% is being used to provide conservatism.	Approved for Use	46
R7	Melted fuel (% of core)	0.375	Westinghouse letter report AEP-99-487. Based on 50% of the rods that violate the DNB limit (15% from R6) having melting in the inner 10% over 50% of the axial length: TBD by Westinghouse. 0.15 x 0.5 x 0.1 x 0.5 = 0.00375 = 0.375%	Approved for Use	46
R8	Gap Fractions I-131 Kr-85 Other noble gases Other halogens Alkali Metals	 0.12 0.15 0.10 0.10 0.10	Per NRC discussion on new source term methodology in DG-1081, Table 3.	Approved for Use	22
R9	Iodine Chemical Species in Containment (%) Elemental Organic Particulate	 4.85 0.15 95.	Consistent with NUREG-1465 containment source terms. DG-1081, Appendix H confirms chemical forms of release to containment atmosphere.	Approved for Use	14 22
R10	Iodine Removal Coefficients in Containment (hr-1) Elemental Organic Particulate	 0 0 0	Westinghouse letter report AEP-99-487. No credit is taken for sedimentation, plateout or containment spray removal of radionuclides in the containment.	Approved for Use	46

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RCCA EJECTION

No.	Design and Licensing Parameters	AEP Prop. Value	Notes	Status	Ref.
R11	Fraction of activity released to primary coolant (for primary to secondary leakage pathway) From gap inventory Iodine Noble Gases From melted fuel Iodine Noble gas	 1.0 1.0 0.5 1.0	Consistent with guidance of Reg. Guide 1.77, Appendix B, paragraph 1.b c "The amount of activity accumulated in the fuel-clad gap should be assumed to be 10% of the iodines and 10% of the noble gases accumulated at the end of core life." Appendix B, paragraph 1.e which says "The ... fraction of the fuel which reaches or exceeds the initiation temperature of fuel melting (typically 2842 Degrees C) at any time during the course of the accident should be calculated, and 100% of the noble gases and 25% of the iodine contained in this fraction should be assumed to be available for release from the containment." Cannot use 25% because it assumes 50% plateout on surfaces. Per Nureg 1465 cannot assume instantaneous plateout. Therefore double 25%.	Approved for Use	11 21
R12	Iodine Chemical Species in Primary Coolant (%) Elemental Organic Particulate	 100 0 0	Standard practice. There is no opportunity to form organic iodine within the primary coolant system. Also, it is conservative to assume that there is no particulate iodine as it would have a higher partition coefficient. DG-1081, Appendix H recommends 97% elemental and 3% organic release via the steam generators. Using 100% elemental is conservative.	Approved for Use	22
R13	Containment Leak Rate (weight %/day) 0-280 Hours > 280 Hours	 0.25 0.125	Containment pressure 25% of limit 280 hours into event per WCAP-14286, Figure 3.5-1. CR-99-12890 identifies condition. 0.25 wt% value is from Tech Spec 3/4.6.1.2. 0.125 wt% value is from Reg Guide 1.4	Approved for Use	13 8
R14	Steam Released to Environment (lbm)	460,063	Westinghouse letter report AEP-99-487. The secondary side release rate is 1.739E6 g/min for 7200 seconds. 453.59 g/lbm conversion factor.	Approved for Use	46
R15	Time to generate signal to switch from CR HVAC Mode 1 to Mode 2 operation (sec)	120	Westinghouse letter report AEP-99-487. An S-signal occures within the first 40 seconds of the event and the emergency HVAC system is conservatively assumed to start 2 minutes after event initiation.	Approved for Use	46
R16	Atmospheric Dispersion Factors for Release from PORVs to Control Room Intake (sec/m^3) 0-2 hr 2-8 hr 8-24 hr 1-4 days 4-5 days 5-30 day	 4.55E-4 2.87E-4 1.23E-4 9.60E-5 6.83E-5 6.83E-5	RD-99-05 is EQE/PLG Calculation C-254055-05. The PORV release point should be used for all intact SG releases. Increase values listed here by 10% to account for partial entrainment in building wake. U2 West PORV to U1 Intake U2 West PORV to U1 Intake U1 East PORV to U1 Intake U1 East PORV to U1 Intake U2 West PORV to U1 Intake U2 West PORV to U1 Intake	Approved for Use	28
R17	Offsite TEDE Dose Acceptance Criteria (Rem)	6.25	25% of LOCA limit per SRP 15.4.8, Appendix A, Section II.	Approved for Use	14

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RCCA EJECTION					
No.	**Design and Licensing Parameters**	**AEP Prop. Value**	**Notes**	**Status**	**Ref.**
R18	Atmospheric Dispersion Factors for Release from Containment Building Surface to Control Room Intake, (sec/m^3)		RD-99-05 is EQE/PLG Calculation C-254055-05	Approved for Use	28
	0-2 Hours	8.95E-4			
	2 to 8 Hours	6.91E-4			
	8 to 24 Hours	2.79E-4			
	1 to 4 Days	2.23E-4			
	4 to 30 Days	2.63E-4			

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Table 1, Dose Conversion Factors, Inhalation

Nuclide Name	Decay Constant (hr^{-1})	CEDE DCF (Rem/Ci)	Gamma Disintegration Energy (Mev/Dis)	Table 2.1	
I-131	0.00359	3.29E4	0.38		
I-132	0.303	3.81E2	2.2		
I-133	0.0333	5.85E3	0.6		
I-134	0.791	1.31E2	2.6		
I-135	0.105	1.23E3	1.4		

Nuclide Name	Decay Constant (hr^{-1})	CEDE DCF (Rem-m^3/Ci-sec)
KR-85M	0.155	0.03707
KR-85	7.37E-6	0.000484
KR-87	0.547	0.146
KR-88	0.248	0.37
XE-131M	0.00241	0.00152
XE-133M	0.0130	0.00553
XE-133	0.00546	0.00624
XE-135M	2.72	0.0775
XE-135	0.0756	0.0482
XE-138	2.93	0.198

Nuclide Name	Decay Constant (hr^{-1})	CEDE DCF (Rem/Ci)
CS-134	3.84E-5	4.62E4
CS-136	2.2E-3	7.33E3
CS-137	2.64E-6	3.19E4
RB-86	1.55E-3	6.63E3
TE-127	7.41E-2	318.
TE-127M	2.65E-4	2.15E4
TE-129M	8.6E-4	2.39E4
TE-129	8.6E-4	90.
TE-131M	2.31E-2	6.4E3
TE-132	8.86E-3	9.44E3
SB-127	7.5E-3	6.04E3
SB-129	0.16	6.44E2
RU-103	7.35E-4	8950.
RU-105	0.156	455.
RU-106	7.84E-5	4.77E5
RH-105	1.96E-2	956.
MO-99	1.05E-2	3960.
TC-99M	0.115	33.
Y-90	1.08E-2	8.44E3
Y-91	4.94E-4	4.89E4

DESIGN INPUT LIST FOR D.C. COOK
OFFSITE & CONTROL ROOM DOSE ANALYSIS

Nuclide Name	Decay Constant (hr^{-1})	CEDE DCF (Rem/Ci)
Y-92	0.196	780.
Y-93	0.0686	2.15E3
NB-95	8.22E-4	5.81E3
ZR-95	4.51E-4	2.37E4
ZR-97	4.1E-2	4.33E3
LA-140	1.72E-2	4.85E3
LA-141	0.176	580.
LA-142	0.45	250.
ND-147	2.63E-3	6.85E3
PR-143	2.13E-3	1.09E4
AM-241	1.83E-7	4.44E8
CM-242	1.77E-4	1.73E7
CM-244	4.37E-6	2.48E8
CE-141	8.89E-4	8.96E3
CE-143	0.021	3.39E3
CE-144	1.02E-4	3.74E5
PU-238	9.02E-7	3.92E8
PU-239	3.29E-9	4.3E8
PU-240	1.21E-8	4.3E8
PU-241	5.5E-6	8.26E6
NP-239	0.0123	2.51E3
SR-89	5.72E-4	4.14E4
SR-90	2.72E-6	1.3E6
SR-91	0.073	1.66E3
SR-92	0.256	8.1E2
BA-139	0.502	1.7E2
BA-140	2.27E-3	3.74E3

ATTACHMENT 7 TO C0600-13

CONTROL ROOM HABITABILITY RESPONSES
to previous concerns on the
Cook Nuclear Plant (CNP) Control Room
from

1. September 15-19, 1986 Survey of the Control Room Ventilation System at CNP Units 1 and 2 performed by Argonne National Laboratory and a member from Nuclear Reactor Regulation's Plant Systems Branch.

2. Unit 1 Licensee Event Report RO 85-007 from March 25, 1985.

3. The Nuclear Regulatory Commission, Nuclear Heating, Ventilation and Air Conditioning Utility Group (NHUG), Nuclear Energy Institute Control Room Habitability Workshop Issues from July 1998.

[#] indicates comment number or section of original document

Control Room Habitability Responses

NRC to AEP dated 02/02/87
September 15-19, 1986 Survey of the Control Room Ventilation System at D.C. Cook Units 1 and 2
Performed by Argonne National Laboratory and a member from NRR's Plant Systems Branch

Item #	Expectations/Suggestions/Comments	Method for Addressing Expectations/Suggestions/Comments
1 [3a]	The present system appears subject to single failure e.g. single normal intake, emergency recirculation, and toilet exhaust dampers and a single Cl_2 detector of each unit.	The control room emergency ventilation system (CREVS) was not designed to meet the single failure criterion. However, damper failures were analyzed and modifications are being completed to add a redundant damper in the normal intake in a series configuration to address the failure of the normal intake air damper to close. A parallel damper is being installed in the emergency/pressurization intake to address the failure of the damper in its normally closed position. To compensate for failure of the recirculation damper to go to the open position, the damper is administratively controlled in a full open position. The toilet exhaust has been permanently removed and the exhaust closed off to prevent a possible source of leakage. Regulatory guides (RGs) 1.78 and 1.95 have been evaluated and the CI_2 detectors are no longer needed for the protection of personnel in the control room for a postulated hazardous chemical release because of the low probability of the event. Revised calculation have been completed and confirm the results of the previous correspondence sent to the NRC on October 11, 1988.
2 [3b]	Interconnection of drains between each air-handling unit and between each reactor's air handling units may present a common mode failure.	A tracer gas test was performed to verify inleakage sources with the existing air handling unit drain line configuration. The measured inleakage flow rate from this test was used in the analysis.

Control Room Habitability Responses

NRC to AEP dated 02/02/87
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Item #	Expectations/Suggestions/Comments	Method for Addressing Expectations/Suggestions/Comments
3 [4]	Technical specifications prepared for the control room may require some changes to be consistent with the Cook III.D.3.4 analysis (See specific comments in Attachment A).	Technical Specification (T/S) changes have been proposed to address the additional operability requirements for the CREVS as recommended in III.D.3.4 of NUREG 0737.
4 [5]	A definitive pressure should be established for the equipment room. An appropriate value might be 1/8-inch water gauge.	The control room envelope (CRE)/pressure boundary has been defined in the T/S Bases and included in Limiting Conditions for Operation (LCO) 3.7.5.1 for ensuring operability of the control room ventilation system. A CRE/pressure boundary definition is proposed for the Bases to include the control room, control room HVAC equipment room, and the plant process computer (PPC) room. The proposed Bases change states that the CRE/pressure boundary shall be considered operable if the rooms can be maintained at a positive 1/16-inch water gauge (WG). The proposed LCO requires maintaining the CRE at a pressure of greater than or equal to a positive 1/16-inch WG consistent with the current design basis.
5 [6]	It does not appear that the toxic gas analysis presented as part of the III.D.3.4 analysis reflects actual system operation.	A new analysis for operation of the control room in the toxic gas mode shows operation is no longer needed as described in an evaluation sent to the NRC on October 11, 1988. The evaluation was completed for possible sources of toxic gas in the surrounding area and also confirmed in a recent evaluation on the Offsite Sources of Toxic Gas and Explosives, reviewed on December 16, 1999. The provisions of RG 1.78 and 1.95 have been evaluated and the toxic gas mode was determined to be no longer necessary for protection of personnel in the control room because of the low probability of the event.

Control Room Habitability Responses

NRC to AEP dated 02/02/87
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6 [7]	Since the SER was issued on III.D.3.4 in 1982 the licensee has altered in at least two instances the amount of filtered makeup flow, the amount of filtered recirculation flow and the amount of assumed unfiltered inleakage into the control room envelope. The licensee's 50.59 evaluations should be reviewed to ensure the 50.59 discussions demonstrated that: (a) the consequences of accidents previously evaluated in the safety analysis report were not increased by the changes made in makeup flow; and (b) the margin of safety as defined in the basis of technical specifications 3/4.7.5.1 was not reduced.	Indiana Michigan Power Company (I&M) has revised the control room dose analysis. The results are addressed in Attachment 1 and a copy of the report is included in Attachment 6. The analysis was conducted using the alternative source term (AST) as defined by NUREG 1465, and demonstrates that there has been no significant increase in the consequences of accidents previously evaluated in the safety analysis report and there is no significant reduction in the margin of safety. Attachment 4 of this submittal describes the basis for these conclusions. A tracer gas test was performed to verify inleakage sources and modifications are being installed to improve the leak tightness of the inlet dampers. The measured inleakage and revised makeup flow rate from this test were used in the analysis.
7 [8]	The training manuals need to be updated to describe the present operating system. Some of the information contained in them is very out-of-date.	The control room ventilation training materials are up-to-date with respect to the as-built system. The Training Department is tracking design changes currently being performed and updating the material as necessary. This is part of the design change process and is an integral part of the release of a design change to other departments.

Control Room Habitability Responses

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8 [B]	A new toxic gas evaluation should be performed based upon the present operating scheme and the associated infiltration rates into the control room envelope through damper leakage, ingress/egress, etc. The capability of the impregnated charcoal to remove the Cl_2 should be evaluated along with the capability of the operator to tolerate this additional quantity of toxic gas. The reanalysis must take into account the time the operator is outside the control room envelope.	A new analysis for operation of the control room in the toxic gas mode shows that it is no longer needed as described in an evaluation sent to the NRC on October 11, 1988. The evaluation was completed for possible sources of toxic gas in the surrounding area and also confirmed in a recent evaluation of the Offsite Sources of Toxic Gas and Explosives, reviewed by plant personnel on December 16, 1999. The provisions of RG 1.78 and 1.95 have been evaluated and the toxic gas mode was determined to be no longer necessary for protection of personnel in the control room because of the low probability of the event.
9 [B]	Self-Containing Breathing Apparati for Control Room Operators is insufficient. The licensee should ensure that a sufficient quantity exists for the operators.	An evaluation was completed for possible sources of toxic gas in the surrounding area. The evaluation determined that there is no longer a need for toxic gas protection for personnel in the control room because of the low probability of the event as described in an evaluation that was sent to the NRC on October 11, 1988.
10 [B]	System parameters for in-leakage and flow rates are different than stated in the III.D.3.4 analysis	Tests have been completed and the revised control room dose analysis uses the most recently verified in-leakage.
11 [B]	III.D.3.4 analysis omitted leakage from ECCS equipment in their calculations of control room operator doses	The control room dose has been revised and includes emergency core cooling system (ECCS) leakage, assumed to begin at the start of the accident equivalent to 0.2 gpm unfiltered ECCS leakage out the unit vent. In addition, a 50 gpm leak from the ECCS piping is assumed to initiate for 30 minutes, 24 hours after entering the recirculation phase.

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12 [C.1]	The designation of the components of the control room emergency ventilation system which shall be OPERABLE should be modified such that technical specification 3.7.5.1.c specifies the emergency filter train and 3.7.5.1.d specifies the control room envelope.	A change to LCO 3.7.5.1 has been proposed to require one High Efficiency Particulate Air (HEPA)/ emergency filter unit and require the CRE/pressure boundary to be operable. The proposed change provides a definition of the filter unit and the CRE/pressure boundary in the T/S Bases. The charcoal adsorber/HEPA filter unit is defined to consist of the prefilter, charcoal adsorber, HEPA filter, and filter housing. The CRE definition proposed for the Bases includes the control room, control room HVAC equipment room, and the PPC room. See discussion in Attachment 1 of this submittal.
13 [C.1]	Air handling units ACRA-1 and ACRA-2 should be included in 3.7.5.1.a (heating and cooling system)	A new T/S 3.7.5.2 has been proposed to relocate the existing temperature control functions described in T/S 3.7.5.1. This new T/S required that the control Room Air Conditioning System (CRACS) be OPERABLE during Modes 1-4.
14 [C.1]	The filter train should include the pre-filter, filter housing, etc. (3.7.5.1.c)	The proposed change to the T/S Bases states that the charcoal adsorber/ HEPA filter unit consists of the pre-filter, charcoal adsorbers, HEPA filter, and filter housing unit, as discussed in Attachment 1.
15 [C.1]	It should be clear that the pressurization fans in item 3.7.5.1.b are emergency filtration system fans ACRF-1A and ACRF-1B for Unit 1 and ACRF-2A and ACRF-2B for Unit 2	A description has been provided in the proposed T/S Bases change that defines two pressurization trains with each pressurization train consisting of a pressurization fan, a normal intake air damper, and an emergency intake air damper available to align and maintain flow to the control room.

Control Room Habitability Responses

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16 [C.1]	The pressure boundary for the GDC-19 doses is the control room envelope and not just the control room. T.S. surveillance 3.7.5.1.d should state "control room envelope."	The CRE has been defined in the proposed T/S change and the operability requirements have been included in the proposed change to LCO 3.7.5.1.c. In the proposed changes to surveillance requirement 4.7.5.1.e.3, the CRE/pressure boundary is verified to be able to maintain the required pressure. The CRE is defined in the proposed change to the Bases to include the control room, control room HVAC equipment room, and PPC room.
17 [C.2]	Action statements should account for the fact there is only one emergency filter train and that limitations on fuel handling are not addressed. For Modes 5-6, must address fuel handling operations with the various portions of the control room emergency ventilation system inoperable.	The proposed changes to T/S 3.7.5.1 expand the applicability of the CREVS. CREVS operability requirement for system operability has been expanded to include Modes 1-4 and during the movement of irradiated fuel assemblies. The proposed T/S change request also includes the action requirements that address fuel handling operations with the various portions of the CREVS inoperable.
18 [C.2]	For modes 1-4 if the filter train is inoperable, then the reactor should be in hot standby in 6 hours, cold shutdown etc. if the filter unit cannot be restored to operable after 24 hours. Seventy-two hours would only be appropriate if the licensee could demonstrate that Unit 2 is capable of maintaining both the Unit 1 and Unit 2 control room envelopes pressurized as required by plant technical specifications and vice versa.	The previous T/S change request to change the allowable outage time to 72 hours was withdrawn. The current T/S has a 24-hour action requirement for an inoperable filter train. If not restored, then the reactor should be in hot standby in 6 hours, cold shutdown in the following 30 hours. These times have been retained in the proposed T/S changes.

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19 [C.3]	T.S. surveillance 4.7.5.1.a should take into account the actual equipment qualification temperature for control room instrumentation and should be measured at the location of the most temperature sensitive equipment.	This concern has been addressed with the issuance of a Safety Evaluation Report (SER) on November 20, 1991, for Unit 1 Amendment No. 159 and Unit 2 Amendment No. 143. The T/S amendment revised the 12-hour T/S surveillance requirement from 120°F to 95°F. The revised temperature was a more realistic temperature based on equipment qualification temperatures.
20 [C.4]	T.S. surveillance 4.7.5.1.b should have system operation occurring for 1 hour. Operation for 1 hour ensures that the system can function without an early trip.	Current T/S surveillance requires the system operate for at least 15 minutes and was accepted by the NRC in a SER dated April 15, 1975. In this SER for Unit 1, discussion item 27 states that surveillance requirements 4.7.5.1 and 4.7.6.1 that were included in the initial issuance of the T/Ss were intended to be applied to plants having heaters upstream of the filters to dry the air entering the filters and is consistent with the recommended testing requirements in NUREG 1431. Donald C. Cook Nuclear Plant (CNP) does not include such heaters. A testing time of 15 minutes is adequate to demonstrate operability of the filter train equipment.
21 [C.5]	The laboratory test for charcoal should be conducted at 30 degrees C utilizing the ASTM D3803 test method and the acceptance criteria should be an allowable penetration of 0.8%.	The proposed T/S change would revise the charcoal testing methodology to ASTM D3803-1989 as recommended by Generic Letter (GL) 99-02. The proposed acceptance criteria for the charcoal adsorbers require the laboratory test to be conducted at 30°C and show a penetration of less than or equal to 1.0% for radioactive methyl iodide. See Attachment 1 for additional discussion.

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22 [C.6]	The in-place DOP and freon test acceptance criteria should be 0.05% penetration for the HEPA filter and charcoal adsorber, respectively consistent with Regulatory Guide 1.52. An allowable penetration of 1% would negate the III.D.3.4 analysis since bypass flow of 1% would negate high efficiency credited for the adsorber and filter.	The assumptions used in the revised control room dose analysis credits an allowable penetration of 1% with a safety factor of 2 applied to the efficiencies. This application of the safety factor is consistent with the recommendations of GL 99-02 of the laboratory carbon samples.
23 [C.7]	T.S. surveillance 4.7.5.1.c.4 and 4.7.5.1.c.5 should verify that the makeup pressurization flow and recirculation flow through the emergency filtration unit total 6000 cfm and are within the bases of Figure 3 of B 3/4.7.5 for unfiltered leakage and filtered flow.	T/S surveillance requirement 4.7.5.1.c.5 and Figure 3 of B 3/4 7-5 were previously removed from the T/Ss. Surveillance 4.7.5.1.c.4 was previously revised to require testing in accordance with ANSI N510-1975 at a system flow rate of 6000 cfm ± 10%. The total system flow rate of filtered and recirculated air are verified to be within the requirements of the surveillance requirement for the design of the pressurization fan.
24 [C.7]	A test should be performed periodically to demonstrate that degradation has not occurred to the system.	The filtered flow rate corresponding to the CRE pressure is measured on a periodic basis (refueling outage frequency) as specified by surveillance requirement 4.7.5.1.e.3.
25 [C.7]	A test should be performed when work is performed on the normal and/or emergency ventilation systems, which could affect the integrity of the system or flow rate.	Current T/Ss require testing "At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system."

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26 [C.8]	T. S. surveillance requirements 4.7.5.1.e.2 should demonstrate automatic initiation of the emergency filtration unit on a high radiation signal.	The revised control room dose analysis does not assume automatic initiation of the CREVS on a high radiation signal as discussed in Attachment 6. The radiation monitors are not powered from an ESF bus and are not considered safety related.
27 [C.9]	Present positive pressure of 1/16 inch W.G. is minimal or a degraded value and should be increased. Measurements made seemed actually to show pressure greater than 1/8 inch W.G.	The plant was originally designed and licensed based on a pressure of 1/16-inch WG. The revised control room dose analysis demonstrates that 1/16-inch WG provides adequate protection.
28 [C.10]	A value for positive pressure in the mechanical equipment room and the computer room should be stated. (4.7.5.1.e.5). Since the air handling units draw over 1300 cfm of supply from this area, and there is a significant in-leakage, the control room would be subject to a significant source of unfiltered air. To minimize this potential, the mechanical equipment room should be made as positive as possible, with a goal of 1/8 inch W.G.	In the proposed T/S change, the CRE is defined in the T/S Bases to include the control room, the control room HVAC equipment room, and PPC room. To be OPERABLE, the CRE must be maintained at a positive ≥1/16-inch WG when compared to the outside atmosphere. The revised control room dose analysis demonstrates that 1/16-inch WG provides adequate protection.
29 [D]	A training instruction was revised in March 1986 and was not at all current. Since this would be one of the first sources an individual would refer to in order to obtain information on the system, the information should be current.	The training materials are up-to-date with respect to the current system design. The Training Department is tracking design changes currently being performed. This is part of the design change process and is integral with the release of a design change to other departments.

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30 [9.1.1]	Procedures 2-OHP-4030.STP.025S, "South Control Room Pressurizer/Cleanup Filter System Operability Test" and 2-OHP-4030.STP.025N, "North Control Room Pressurizer/Cleanup Filter System Operability Test" should have the title changed and any references to the fan changed from north and south to east and west or the fan numbers ACRF-1 and ACRF-2 should be used as labeled on P&ID OP-2-15149.	The procedures now correctly identify the fans as unit specific east and west fans. For the east control room pressurization fan, the procedure correctly identifies the fan as 1-HV-ACRF-1 and the west fan is identified as 1-HV-ACRF-2. Procedures 2-OHP-4030.STP.025S and 2-OHP-4030.STP.025N have been revised to 1/2-OHP-4030.STP.024E, "East Control Room Pressurizer/Cleanup Filter System Operability Test" and 1/2-OHP-4030.STP.024W, "West Control Room Pressurizer/ Cleanup Filter System Operability Test."
31 [9.1.2]	In the procedures in item 9.1.1 (above) and the applicable procedures for Unit 1, it would seem to be a good engineering practice to record filter train differential pressures when the filter train is tested.	HEPA and charcoal filters differential pressure are measured and recorded as part of the procedure requirements.
32 [9.1.3]	Procedure 1-OHP 4021-028-015, "Operation of the Control Room Pressurizer/Cleanup Filter System," references procedures 1-OHP.4030-STP-024N and 024S. The referenced procedures should be 024E and 024W.	The procedure is now numbered 1/2-OHP-4021.028.014, "Operation of the Control Room Pressurizer/Cleanup Filter System." The current procedure contains no reference to the system operability test procedures for the pressurization fans.
33 [9.1.4]	Procedures (**) 12-THP 4030 STP-229 should have a step that requires adequate levels of lighting when making the visual inspection.	In section 5.3 of the procedure, a step requires inspection of the ventilation housing for adequate lighting to perform the visual examination of housing and components by verifying the bulbs are not burned out.

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34 [9.1.5]	Procedure (**) 1 THP 6040 PER.094, "Control Room Ventilation Balancing," should have an acceptance criteria for the quantity of outside air makeup.	The procedure for both Unit 1 and Unit 2 is being revised to verify makeup air flow is within the acceptance criteria.
35 [9.2.1]	The computer room communicates with the HVAC equipment room via grated louvers. There is a fire damper between the louvers but isolation requires heat to melt a fusible link which allows the fire damper to close. The fire damper appears to be the louvered or venetian blind type that have demonstrated a propensity of "hanging up" on an edge when they are released after the fusible link melts. Thus, in the event of actuation of the computer room CO(2) system, CO(2) will probably be circulated throughout the control room envelope due to the failure of the fire damper to isolate the computer room.	Fire dampers HV-ACFD-1, 2, and 3, located between the PPC rooms and the control room HVAC equipment rooms for the Unit 1 and Unit 2 control rooms, are tested for closure by a halon system actuation test on a one year frequency. In addition, the dampers are tested for closure from the thermal link heat activation under a separate frequency. Damper closure is the test acceptance criterion. Failure to close will be documented in a condition report and remedied by the work control process. No CO_2 system is in service for the PPC rooms. Therefore, no CO_2 has any possibility to enter the control rooms. Halon, from a low concentration system and non-toxic at this concentration level, is only used in non-safety related areas, i.e. PPC rooms.

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36 [9.2.2]	CNP is staffed by six people per unit. There are two emergency breathing units in the control room with 10 additional units immediately outside the control room for a total of 12 units readily available. Reg. Guide 1.95 requires that there be one extra emergency air breathing unit for every three units required. Reg. Guide 1.95 also requires the emergency air breathing units be donned and in use within two minutes of receiving a chlorine alarm. It is doubtful that operators could don the air breathing units in two minutes if the units are outside the control room. Therefore, a total of 16 units should be available to the control room.	A new analysis for operation of the control room in the toxic gas mode is no longer needed as described in an evaluation sent to the staff on October 11, 1988. The evaluation was completed for possible sources of toxic gas in the surrounding area and confirmed in a recent evaluation performed by I&M on the Offsite Sources of Toxic Gas and Explosives, reviewed on December 16, 1999. The provisions of RG 1.78 and 1.95 have been evaluated and the toxic gas mode was determined to be no longer necessary for protection of personnel in the control room because of the low probability of the event.

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37 [9.2.3]	The FSAR Section 9.10.2 indicates that charcoal adsorbers have a removal efficiency of 99.9% of entrained methyl iodide or iodine vapor and Table 14.3.5-9 indicates a removal efficiency of 95% for the same charcoal adsorber. If a charcoal efficiency of 95% is to be used in the calculations for control room operator exposure for a DBA, the charcoal must be tested for an efficiency of 99.3%. Using a test criteria of 90% as required by Technical Specification would require a value of 30% be used for the dose calculation. The dose calculations following a DBA should be reevaluated based on allowable charcoal efficiencies based on actual test specifications.	For methyl idoide testing of charcoal adsorbers, the proposed T/S specify a penetration of less than or equal to 1% with testing performed in accordance with ASTM D3803-1989. This reflects a safety factor of 2 consistent with the guidance provided in GL 99-02.
38 [9.2.4]	The outside air intakes for the air handling units should have two isolation valves in series to meet the single failure criteria.	The control room emergency ventilation CREVS was not designed to meet the single failure criterion. However, modifications are being completed on the normal intake to install new bubble tight dampers. The existing damper will be replaced and a redundant damper installed in the normal intake in a series configuration.

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39 [9.2.4]	Reg. Guide 1.95 requires that the Cl(2) detectors cause an automatic isolation of the control room within 10 seconds of sensing chlorine in the duct and that the Cl(2) be sensed by two physically separated channels for each fresh air inlet. Cook has only one channel on the normal outside air inlet in each unit and the Cl(2) detectors actuates alarms with isolation by manual operation.	A new analysis for operation of the control room in the toxic gas mode shows it is no longer needed as described in previous correspondence with the NRC. RG 1.95 has been evaluated and it has been determined that the Cl_2 detectors are no longer needed for protection of personnel in the control room because of the low probability of the event. The revised calculations confirm the results of the previous correspondence.
40 [9.3.1]	In Technical Specification 4.7.5.1.e.3, the HVAC Mechanical Equipment Room is part of the control room envelope, it should be maintained at 1/8-inch W.G. Delta P to surrounding areas.	In the proposed T/S change, the CRE is defined in the T/S Bases to include the control room, the control room HVAC equipment room, and PPC room. To be OPERABLE, the CRE must be maintained at a positive ≥1/16-inch WG when compared to the outside atmosphere. The revised control room dose analysis demonstrates that 1/16-inch WG provides adequate protection.
41 [9.3.2]	In Technical Specifications 4.7.5.1.c.3, 4.7.5.1.d.1 and 4.7.5.1.d.2, laboratory test for charcoal does not have the correct acceptance criteria. If 95% efficiency is assumed for the safety analysis, the laboratory test should demonstrate an efficiency of >=99.3% (<=.07% penetration).	For methyl iodide testing of charcoal adsorbers, the proposed T/S specify a penetration of less than or equal to 1% with testing performed in accordance with ASTM D3803-1989. This reflects a safety factor of 2 consistent with the guidance provided in GL 99-02.

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42 [9.3.3]	In Technical Specification 4.7.5.1.c.3, the laboratory test conditions for charcoal should be specified at 30 degrees C, 95% R.H. Test method ASTM D 3803 should be specified.	The proposed change revises the charcoal testing methodology to ASTM D3803-1989 as recommended by GL 99-02. The proposed surveillance requirements for T/S 4.7.5.1.c.3, 4.7.5.1.d.1, and 4.7.5.1.d.2 for the charcoal adsorbers require the laboratory test to be conducted at 30°C, 95% RH, and show a penetration of less than or equal to 1.0% for radioactive methyl iodide in accordance with ASTM D3803-1989.
43 [9.3.4]	Specification 4.7.5.1.a. requires once every 12 hours the control room temperature be verified <=120 degrees F. A more realistic temperature should be selected based on equipment qualification temperatures and a temperature survey to determine where the hottest areas of the control room are and what effect those hot spots have on instrumentation.	In an SER issued to I&M on November 20, 1991, for Amendment No. 159 to the Unit 1 T/Ss and Amendment No. 143 to the Unit 2 T/Ss, the NRC accepted the control room temperature survey results and approved the reduction in the allowable control room temperature from less than or equal to 120°F to less than or equal to 95°F.
44 [9.3.5]	The inplace test for HEPA filter and/or charcoal adsorbers in Specification 4.7.5.1.c.1, 4.7.5.1.c.2, 4.7.5.1.d.2, 4.7.5.1.f, and 4.7.5.1.g should specify <0.05% penetration per ANSI N509-1980.	For methyl iodide testing of charcoal adsorbers, the proposed T/S specify a penetration of less than or equal to 1% with testing performed in accordance with ASTM D3803-1989. This results in a safety factor of 2, which consistent with the guidance provided in GL 99-02. For testing of HEPA filters, the existing T/S specify an efficiency of greater than or equal to 99% with testing performed in accordance with ANSI N510-1975. This is greater than the 98% efficiency assumed in the revised control room dose analysis.

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45 [9.3.6]	In Specification 4.7.5.1.c.2, an additional step should be added to verify isolation on a high radiation signal from the control room area monitor.	The revised control room dose analysis does not assume automatic initiation of the CREVS on a high radiation signal as discussed in Attachment 6. The radiation monitors are not powered from an engineered safety features (ESF) bus and are not considered safety related.
46 [9.4.2]	(1) There is [SIC] significant amounts of inleakage from the outside air intakes through the charcoal adsorbers and into the control room envelope during the toxic gas mode of operation.	For Unit 2, modifications have been completed on the normal intake to install new bubble tight dampers. The existing damper has been replaced and a redundant damper will be installed in a series in the normal intake air duct. Inleakage tests have been completed and the modifications on the normal intake have significantly reduced inleakage through the dampers. Similar modifications and testing are planned for Unit 1. The toxic gas mode is no longer needed as described in an evaluation sent to the staff on October 11, 1988. The evaluation was completed for possible sources of toxic gas in the surrounding area and confirmed in a recent evaluation of the Offsite Sources of Toxic Gas and Explosives. The provisions of RG 1.78 and 1.95 have been evaluated and the toxic gas mode was determined to be no longer necessary for protection of personnel in the control room because of the low probability of the event.

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47 [9.4.2]	(2) There is a significant inleakage into the Unit 2 emergency filter train when the system is operating in the emergency mode of operation. The outlet flow was as much as 3000 cfm greater than the inlet. This would suggest that the installed flow measuring device can only reflect flow in the filter trains supply duct and may not be showing all the flow in the filter housing which exists due to filter housing leaks. It would seem appropriate to install a flow-measuring device in the outlet ductwork.	The flow-measuring device has been installed in the Unit 1 and 2 pressurization train outlet ducts.

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48 [9.4.2]	(8) Smoke test showed that air flowed into or out of the condensate drain lines for the air handling units. This condition existed in both Unit 1 and Unit 2 and the direction of flow varied depending upon which pieces of equipment were running. If a proper loop seal existed on the drain lines, there would be no airflow. Station personnel explained that drain piping from both equipment rooms are joined together and then go to a common loop seal. The condensate drain lines connect the two control rooms together. Therefore, if the integrity of one control room envelope is broken the other control room envelope is also broken.	A tracer gas test was performed to verify inleakage sources with the existing air handling unit drain line configuration. The measured inleakage flow rate from this test was used in the analysis.

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49 [9.4.2]	(9) The discharge of the air handling units in the Unit 2 control room HVAC system has a consistently lower flow than the inlet to the AHU's. The combination of branches (control room and computer room supplies) downstream of the AHU's discharge is always higher. The utility should investigate this condition and determine why the flow at points 5 and 6 are lower than the downstream combination of points 7 and 8 (see Table 2 and Table 4).	After reviewing the air flows determined by Argonne National Laboratory, it appears both the AHU inlet (return) air flows and supply airflow to the control room and computer room are in agreement with each other. The AHU discharge (or outlet) air flows are not in agreement with the return and control room/computer room air flows. It appears the discharge air flows were measured via a set of flow ports located just downstream of a duct heater, and just upstream of a back draft damper. With these two disturbances, it is possible that some of the readings in the flow traverse were taken in spots with "low" velocities (caused by the disturbances). This would ultimately lower the average velocity and airflow at that location. As stated in Item 34, the flow balance procedure is to be performed again after installation of the new dampers for the control room AHUs.

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50 [9.5]	Although the control room is being pressurized to greater than 0.125 in. H(2)O, credit is only taken for 0.0625 in. H(2)O. With this assumption, a value higher than 10 cfm should be used for unfiltered air inleakage due to ingress-egress. Assuming 20 cfm unfiltered inleakage due to ingress-egress and 40 cfm inleakage across isolation dampers, the chart used for thyroid dose is at the upper limit of 60 cfm unfiltered inleakage. With 60 cfm unfiltered inleakage and 1100 cfm filtered outside air makeup, there is not much margin before the GDC-19 limit of 30 rem thyroid is reached.	I&M has revised the control room dose analysis and the results are discussed in Attachment 1 and a copy of the report is included in Attachment 6. The analysis was conducted using the AST as defined by NUREG 1465, and the results meet the GDC-19 limits as described in 10 CFR 50.67. A tracer gas test was performed on Unit 2 to verify inleakage sources and modifications are being installed to improve the leak tightness of the inlet dampers. During the performance of the tracer gas testing, the control room pressure was .113-inch WG as compared with atmospheric pressure. Filtered outside air makeup was measured at 631 ± 33 standard cfm. The test measured unfiltered inleakage of 49 ± 49 cfm with a 95% confidence limit on the upper and lower bounds of these values on Unit 2. The measured inleakage during the tracer gas testing accounts for normal ingress and egress into the control room. The revised control room dose analysis includes a makeup flow rate of 1000 cfm with an unfiltered inleakage at the upper bounding limit of 98 cfm. The Unit 2 test is expected to bound the Unit 1 results following installation of redundant intake dampers on Unit 1.
51 [9.6]	There were no LER's associated with the loss of cooling to the control room envelope. The AHU capacity for both units seem to be more than adequate. However, the Technical Specification Limit of 120 degrees F needs further evaluation (see Item 8.4.4).	In an SER issued to I&M on November 20, 1991, for Amendment No. 159 to the Unit 1 T/Ss and Amendment No. 143 to the Unit 2 T/Ss, the allowable control room temperature was revised from less than or equal to 120°F to less than or equal to 95°F.

Date: March 25, 1985
Unit 1 Licensee Event Report RO 85-007

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52	To prevent recurrence, modifications are being developed to improve habitability and operability of the Control Room HVAC System.	I&M has revised the control room dose analysis and the results are addressed in Attachment 1 and a copy of the report is included in Attachment 6. The analysis was conducted using the AST as defined by NUREG-1465, and demonstrates the results meet the proposed GDC-19 limits as described in 10 CFR 50.67. The damper concerns as discussed in the Licensee Event Report (LER) were analyzed and modifications are being installed to add a redundant damper to normal intake in a series configuration to address the failure of the normal intake air damper to close. A parallel damper is being installed in the emergency/pressurization intake to address the failure of the damper in its normally closed position. To compensate for failure of the recirculation damper to go to the open position, the damper will be administratively controlled in a full open position. The toilet exhaust has been permanently removed and the exhaust closed off to prevent a possible source of leakage.
53	Bubble-Tight dampers will be provided in the normal outdoor intake.	Two bubble-tight dampers are being installed in series in the normal outside air intake duct.
54	The return dampers from the mechanical equipment room will be replaced with heavy-duty dampers, which will facilitate damper balancing.	The return damper from the HVAC equipment room is being changed as described in the LER.
55	An airflow monitoring station will be provided in the cleanup/pressurization system to allow quick and accurate system flow measurement and adjustment.	Flow test points on the suction and discharge side of the pressurization system have been provided.

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56	A permanent seal will be provided for the floor and equipment drain system.	A permanent loop seal is installed in the common drain line connecting the drains from Unit 1 and Unit 2 control room HVAC equipment rooms to the turbine building drain system.
57	A continuing effort will be made to seal existing penetrations.	The penetrations were sealed to reduce the inleakage into the CRE. The tracer gas testing performed in June 1999 identified the amount of unfiltered inleakage into the CRE and these results were used as a verified input into the revised control room dose analysis.
58	A technical specification change request will be submitted which should more concisely reflect the requirements in both the pressurization and recirculation modes.	A change to LCO 3.7.5.1 has been proposed to separate each individual function. The proposed T/S LCO 3.7.5.1.b requires the pressurization fans and associated dampers to be operable as part of a "pressurization train" and a definition has been added to the CREVS bases. The proposed definition of a pressurization train includes the pressurization fan, normal air intake damper, and emergency air intake damper for the associated electrical division. A new T/S 3.7.5.2 has been proposed to relocate the existing temperature control functions described in T/S 3.7.5.1. This new T/S required that the Control Room Air conditioning System (CRACS) be OPERABLE during Modes 1-4.

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59	Because the reanalysis assumed 95% charcoal filter efficiency for methyl iodides, administrative controls will be initiated prior to next scheduled surveillance to establish 95% as the minimum test acceptance criteria.	The proposed T/S change request included in this submittal revises the efficiency requirement to meet the 95% assumptions of the revised control room dose analysis. The proposed surveillance requirements for the charcoal adsorbers require the laboratory analysis to show a penetration of less than or equal to 1.0% for radioactive methyl iodide or a 99% charcoal filter efficiency for removal of the radioactive methyl iodides. See Attachment 1 for more discussion on the proposed changes.
60	The skin dose was initially calculated by a rationing technique as 73 rem. Because this was above NUREG 0737 limits of 30 rem, the plant manager was notified that operators would require protective clothing at some point during the accident as determined by health physics personnel. Subsequent reanalysis by Westinghouse confirmed that the 74 rem value was overly conservative. The more refined Westinghouse analysis determined values of 12.0 rem based on a conservative 800 cfm of inleakage. The Westinghouse calculation is undergoing review. The recommendation for protective clothing will be withdrawn when the calculations are deemed acceptable.	I&M has revised the control room dose analysis and the results are included in Attachment 6. The analysis was conducted using the AST and demonstrates a control room dose of ≤ 5 rem Total Effective Dose Equivalent (TEDE) for all the accidents analyzed. The results indicate that dose to the control room personnel meets the criterion expressed in GDC-19 as proposed in 10 CFR 50.67. Requirements for protective clothing will be withdrawn following NRC approval to implement the AST.

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61	Control room envelope inleakage >> leakage assumed in licensing analysis.	I&M has revised the control room dose analysis and the results are described in Attachment 1 and a copy of the report is included in Attachment 6. The analysis included the results of a tracer gas test performed to verify inleakage sources. The measured inleakage and revised makeup flow rate from this test were used in the analysis. Modifications are being completed on the normal intake and the emergency intake to reduce inleakage through the dampers. The toilet exhaust has been permanently removed and the exhaust closed off to prevent another possible source of leakage.
62	Control room envelope and ESF ventilation system constructed and / or operated inconsistent with the licensing basis.	A change to LCO 3.7.5.1 has been proposed to require both pressurization trains, one HEPA/emergency filter unit, and the CRE/ pressure boundary to be operable. The proposed change provides a definition of a pressurization train, the filter unit, and the CRE/pressure boundary in the T/S Bases. The proposed T/S change revises the charcoal testing requirements for the CREVS and the ESF ventilation systems to meet the recommendations of GL 99-02.
63	Licensees rely on compensatory actions such as SCBAs and / or KI pills to mitigate the consequences.	I&M has revised the control room dose analysis and the results are included in the submittal and described in Attachment 1 and Attachment 6. The analysis will not require the use of compensatory actions to provide personnel protection once the analysis is approved. The analysis was conducted using the AST and the results indicate that dose to the control room personnel meets the 10 CFR 50.67 limits of fewer than or equal to 5 rem TEDE.

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64	Some Licensees still owe the staff analyses.	The results of a revised control room dose analysis are included in this submittal and described in Attachment 1. A copy of the report with the verified inputs for all assumptions is included in Attachment 6. The analysis was conducted using the AST and demonstrates that dose to the control room personnel meets the 10 CFR 50.67 limits of ≤ 5 rem TEDE.
65	Power Uprates, ARC, SG Replacements ignore control room habitability.	The revised control room dose analysis conservatively includes a power uprate for each unit and other changes to the assumed inputs as described in Attachment B of the analysis. The analysis is provided as Attachment 6 to this submittal.
66	Accidents considered are too limited.	For the revised control room dose analysis, loss-of-coolant accident (LOCA) and non-LOCA events were evaluated to determine the limiting accident for the contribution of dose to control room personnel. The results of this analysis are discussed in Attachment 1 and Attachment 6 of this change request.
67	Licensee with an 8600 gal. tank of NH_3 had no procedures for combating a spill including alignment of ventilation systems and the need for SCBAs.	I&M has implemented a HAZMAT program which includes procedures that specify the actions required to combat and contain a hazardous material spill.

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68	Failure to recognize that source term mix vary [SIC] between accidents and that radiation monitor response to such mixes will also vary.	For the revised control room dose analysis, the LOCA and non-LOCA events were evaluated to determine the limiting accident for the contribution of dose to control room personnel. The analysis was conducted using the AST as defined by NUREG-1465. No credit was taken for the initiation of the emergency mode of operation for any of the CNP ventilation systems by a radiation monitoring system. The result of this analysis is discussed in Attachment 1 and Attachment 6 of this change request.
69	Failure to incorporate into the dose assessment the activity from normal supply air prior to its isolation.	Assumptions for the activity from the normal air intake prior to the accident were addressed in the revised control room dose analysis. The results of the analysis are described in Attachment 1 and in Attachment 6. All inputs to the analysis have been verified to update the assumptions used for each accident analyzed.
70	Licensees believe GDC 19 is limited to LOCA analysis	For the revised control room dose analysis, LOCA and non-LOCA events were evaluated to determine the limiting accident for the contribution of dose to control room personnel. The revised control room habitability analysis report is included in Attachment 6 of this change request and identifies that the rod ejection accident is the most limiting using the AST.

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71	Licensees have system engineers unaware of licensing basis.	During the current outage, engineering support personnel (ESP) training classes are being conducted that cover the requirements and inter-relationship of design, operation, and licensing requirements for plant systems. The system engineers have completed orientation training, engineering system readiness reviews, and licensing basis review courses.
72	Licensees make changes to design and operation without regard for licensing consequences.	Quarterly ESP training classes are developed with the assistance of a Curriculum Review Committee. The items selected for the training program cover the requirements and inter-relationships of design, operation, and licensing as applicable for the specific system.
73	Licensees are dependent upon outside organizations for accident analyses.	I&M has adequate in-house expertise. However, outside organizations may be used when there are advantages in using expertise from vendors.
74	Technical Specifications weakness: performance of ESF ventilation system and control room isolation (not?) addressed.	The proposed changes to T/S 3.7.5.1 expand the applicability of the CREVS operability requirements to include Modes 1-4 and during the movement of irradiated fuel assemblies. The proposed T/S change also includes action requirements that address fuel handling operations with the various portions of the CREVS inoperable. The proposed changes to LCO 3.7.5.1 include the operability requirements for the HEPA/ emergency filter unit and the CRE, which is defined in the T/S Bases. GL 99-02 recommendations on charcoal testing are being included in the ventilation system requirements.

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75	Technical Specifications weakness: no periodic surveillance of envelope integrity.	In the proposed changes, the CRE has been defined and operability requirements included in LCO 3.7.5.1.c. A makeup flow rate of 1000 cfm is proposed in the revision to the associated surveillance requirement. The proposed CRE definition in the bases includes the control room, control room HVAC equipment room, and the PPC room. The surveillance requires demonstrating that the CRE maintain a positive 1/16-inch WG pressure.
76	Analyses that only evaluate LOCAs, when other accidents may be limiting. Don't assume LOCA is limiting.	For the revised control room dose analysis, LOCA and non-LOCA events were evaluated to determine the limiting accident for the contribution of dose to control room personnel. The revised control room habitability analysis report is included in Attachment 6 of this change request and identifies that the rod ejection accident is the most limiting using the AST.
77	Analyses performed at existing plant conditions rather than at Technical Specification or design values.	All inputs to the revised control room dose analysis were verified to update the assumptions used for each accident analyzed. The inputs used in the analysis are included in the second part of Attachment 6.

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78	Unrecognized and unaccounted for differences in control room isolation and control room ESF ventilation system responses as a function of the accident (e.g., different release points). Do recognize control room isolation and control room ESF ventilation system responses may vary with the accident.	All inputs to the revised control room dose analysis have been verified to update the assumptions used for each accident analyzed. LOCA and non-LOCA events were evaluated to determine the limiting accident for the contribution of dose to control room personnel. Accidents that do not assume the CREVS to isolate until directed by the plant procedures, such as fuel handling accident and loss of offsite power, have been addressed in Attachment 6. No credit was taken for the initiation of the emergency mode of operation for any of the CNP ventilation systems by a radiation monitoring system.
79	Lack of release location specific Chi/Q values. Don't assume that one χ/Q represents all accidents' release pathways.	I&M has revised the control room dose analysis and the results are included in Attachment 6. The report contains reanalyzed points of origin for the atmospheric dispersion factors (χ/Q), using the most limiting value for the event.
80	Accidents from adjacent unit(s) not addressed. Assess the impact of adjacent units.	Considering the effects of an accident on the adjacent unit is not part of the CNP current licensing basis and design basis. 10 CFR 50.36 criteria do not require T/S to address possible effects of an accident on the adjacent unit.
81	Inability of radiation monitors to isolate normal ventilation and initiate ESF ventilation.	T/S surveillance requirement 4.7.5.1.e.2 does not require automatic initiation of emergency filtration unit on a high radiation signal since the radiation monitors are not safety related. No credit was taken for the initiation of the emergency mode of operation for any of the CNP ventilation systems by a radiation monitoring system.

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82	Inappropriate assumptions of charcoal adsorber efficiency. Don't assume the charcoal adsorber efficiency is the same as the Technical Specification value for the laboratory test.	For methyl iodide testing of charcoal adsorbers, the proposed T/S specify a penetration of less than or equal to 1% with testing performed in accordance with ASTM D3803-1989. This results in a safety factor of 2, which is consistent with the guidance provided in GL 99-02.
83	Dose assessments which fail to incorporate delays in control room isolation and ESF ventilation filter actuation due to requirements for operator actions and/or loss of offsite power. Factor into your analysis delays in control room isolation and ESF ventilation filter actuation due to LOOP.	The accidents that do not result in control room ventilation isolation until directed by the plant procedures have been addressed in the revised control room dose analysis. LOCA and non-LOCA events were evaluated to determine the limiting accident for the contribution of dose to control room personnel. All inputs to the analysis have been verified to update the assumptions used for each accident analyzed. A loss of offsite power would not be expected to result in a safety injection signal; therefore, the automatic actuation of control room HVAC from the normal operation mode to the accident mode of operation would only occur from a high radiation monitor signal. Since the high radiation monitor is not safety grade, this system is not assumed to operate in the analysis and the control room HVAC system is not assumed to switch to the emergency mode. The analysis assumes that the control room HVAC system remains in the normal operational mode with 1000 cfm of unfiltered makeup air flow drawn into the control room for the duration of the event.

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84	Utilization of non-safety grade systems and/or components to mitigate the consequences. Don't assume credit for non-safety grade equipment and/or components.	No credit was taken for non-safety grade equipment or components for any of the revised analyses. No credit was taken for the initiation of the emergency mode of operation CREVS and storage pool ventilation systems by a radiation monitoring system since they are not considered safety grade.
85	Don't assume your control room inleakage is limited to ingress/egress.	A tracer gas test was performed to verify inleakage sources and modifications are being installed to improve the leak tightness of the inlet dampers. The measured inleakage and revised makeup flow rate from this test were used in the analysis. The measured inleakage during the tracer gas testing accounts for other sources of inleakage in addition to normal ingress and egress into the control room.
86	Incorporate all appropriate pathways into the analysis (e.g. MSIV, ECCS, containment, and bypass).	The revised control room dose analysis analyzed the various accidents and revised verified inputs. The revised inputs include revised inleakage values in the auxiliary building and reanalyzed points of origin for the χ/Q. The LOCA and non-LOCA events were evaluated to determine the limiting accident for the contribution of dose to control room personnel and included all appropriate pathways.
87	Don't presume that maintaining a positive pressure in the control room is an indication of integrity.	The CRE is maintained at 1/16-inch WG in the pressurization mode of operation to minimize inleakage. A tracer gas test demonstrated inleakage sources to the CRE. The proposed T/S change in LCO 3.7.5.1.c requires the CRE to be operable and to maintain the pressure boundary at a positive pressure of greater than or equal to 1/16-inch WG.

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88	Incorporate Technical Specifications values into the analysis.	The revised control room dose analysis includes the proposed T/S values in the assumptions used in Attachment 6. The T/S values for charcoal efficiencies, makeup air flow rate, and pressures are included in the verified inputs for the analysis. The revised control room dose analysis used 95% as the efficiency for iodine removal for organic and elemental iodines and 98% efficiency for particulate iodine removal.
89	NUREG/CR-5055: Should be in accordance with accepted theory and practice. Meander and building wake adjustments should be treated separately.	The revised control room dose analysis contains reanalyzed points of origin for the χ/Q factors, using the recommended ARCON96 computer program to determine the most limiting value for each event. Meander and building wake adjustments were treated separately. The most limiting χ/Q factors were used for all the analyzed accidents and their contribution to the control room dose.
90	ARCON96 Reviews being performed on a case-by-case basis. Sensitivity study currently underway.	New analysis included the use of ARCON96 computer code to determine the χ/Q values.
91	Data Representativeness: Overall site Long term conditions System maintenance	The information supplied in this amendment is the latest data available to I&M for the revised control room dose analysis. The proposed T/S changes are based on the latest changes to the associated system design. The input parameters for the revised control room dose analysis represent the latest information available from the results of tests performed for verification of data.

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92	Submittals: More detail needed in amendment request submittals. Methodology, inputs, assumptions, and the bases for selection.	I&M has provided the revised control room dose analysis in Attachment 6. The report contains the information on the CNP control room dose, the various accidents analyzed, and changes used in the analyses inputs, such as revised inleakage values and reanalyzed points of origin for the χ/Q. The LOCA and non-LOCA events were evaluated to determine the limiting accident for the contribution of dose to control room personnel. The analyzed accidents were in accordance with the recommendations of NUREG 1465 for use of the AST.
93	Tracer Concentration Decay Test: Works best for lower makeup rates. Provides total air inflow into CRE. Must measure makeup to obtain air inleakage. Must measure volume of CRE. For systems with no makeup, directly provides air inleakage.	I&M did not use this test. For the new tracer gas test, I&M used a Concentration Buildup/Steady State Test to demonstrate the control room inleakage. Makeup air flow rate was measured during the tracer gas testing.
94	Tracer Concentration Decay Test- Test Conditions: Takes 4 to 8 hours, lower air inleakage rates take longer. Can't change ventilation lineup during test. Must ensure that tracer permeates entire CRE, concentration must be approximately the same everywhere in CRE. Common test errors usually result in higher inleakage values.	I&M did not use this test but instead used the Concentration Buildup/ Steady State Test.

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Item #	Expectations/Suggestions/Comments	Method for Addressing Expectations/Suggestions/Comments
95	Concentration Buildup / Steady State Test: Works best for higher makeup rates. Provides total air inflow into CRE. Must measure makeup to obtain air inleakage. Often can simultaneously measure makeup using tracer technique. Measured inleakage is independent of CRE volume.	Two different tests were performed on the CRE. The first test was a simple damper test that measured the leakage past the makeup airflow isolation damper on the CRACS. The second test was a measurement of the overall unfiltered inleakage into the CRE. This test was a constant injection test (concentration build-up/steady state). The test met all of the items recommended in the NHUG table. In addition, the makeup air flow rate was measured during the tracer gas testing and the CRE/ pressure boundary differential pressures were verified.
96	Concentration Buildup / Steady State Test- Test Conditions: Takes 6 to 12 hours- must wait for concentration equilibrium. Can't change ventilation lineup during test. Must ensure that tracer penetrates entire CRE. CRE must be well mixed. Common test errors usually result in higher inleakage values.	The tracer gas test was a constant injection test and met all of the items listed in the NHUG table for "Concentration Build-up/Steady State Test-Test Conditions. In addition, the makeup air flow rate was measured during the tracer gas testing and the CRE/pressure boundary differential pressures were verified.

Date: July 1998
NRC, Nuclear HVAC Utility Group (NHUG), NEI Control Room Habitability Workshop Issues

Item #	Expectations/Suggestions/Comments	Method for Addressing Expectations/Suggestions/Comments
97	Tracer Gas Measurement System Requirements: All parts of measurement system must be well characterized. Must use recognized and appropriate instrumental techniques. NIST traceable, "tight" calibrations for flow measurement systems, tracer analyzer concentration standards, and tracer injection gas concentrations.	The tracer gas test was set up and performed in accordance with the following ASTM standards: E260-91 "Standard Practice for Packed Columns Gas Chromatography," E697-91 "Standard Practice for Use of Electron Capture Detectors in Gas Chromatography," and E741-93 "Standard Test Method for Determining Air Change Rate in a Single Zone by Means of a Tracer Dilution." The instrumentation, calibration, and injection gases used during the test were all traceable to the required standards.

Date: July 1998
NRC, Nuclear HVAC Utility Group (NHUG), NEI Control Room Habitability Workshop Issues

Item #	Expectations/Suggestions/Comments	Method for Addressing Expectations/Suggestions/Comments
98	Possible resolution for control room habitability: 1. Determination of control room envelope integrity. 2. Re-assessment of appropriateness of accident assumptions. 3. Verification of control room and control room ventilation systems design and operation relative to the licensing basis. 4. Incorporation into Technical Specifications requirements for periodic demonstration of control room envelope integrity. 5. Equivalent organ dose = 50 rem thyroid. 6. Closeout of TMI III.D.3.4 remnants.	I&M has revised the Control Room Dose Analysis and the results are included in Attachment 1 and in Attachment 6. The report contains the information on the CNP control room dose, the various accidents analyzed, and changes used in the analyses inputs, such as revised inleakage values and reanalyzed points of origin for the χ/Q. A tracer gas test was performed to measure the inleakage, makeup flow rate, and CRE pressure relative to the adjacent areas and outside atmospheric pressure. The analysis evaluated the LOCA and non-LOCA events to determine the limiting accident for the contribution of dose to control room personnel. Modifications are being installed to improve the leak tightness of the inlet dampers. A proposed revision to the T/Ss adds new operability requirements, actions, and surveillances to address the CRE and new components of the system. As required by NUREG 0737, "Clarification of TMI Action Plan Requirements," III.D.3.4 for control room habitability requirements, the operability requirements for the CREVS have been expanded to include operations involving the movement of irradiated fuel assemblies. All results indicate that dose to the control room personnel is less than the 10 CFR 50.67 limit of 5 rem TEDE.
99	Risked Based Arguments: Old design vs. new design	A risk analysis was conducted for the proposed use of AST, the proposed changes to the T/Ss, and for the system modifications to be installed in the CREVS. This analysis is presented in Attachment 1.

ATTACHMENT 8 TO C0600-13

EVALUATION OF THE POST-LOCA EQUIPMENT QUALIFICATION DOSE

Westinghouse
Electric Company LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

AEP-00-077
March 02, 2000

Mr. Jeb Kingseed
American Electric Power
500 Circle Drive
Buchanan, Michigan 49107

AMERICAN ELECTRIC POWER
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
Equipment Qualification Dose Report

Dear Mr. Kingseed,

Attached for AEP use is a report documenting an evaluation of the Donald C. Cook Nuclear Plant post-LOCA equipment qualification (EQ) dose. The information provided is based on the NUREG-1465 source terms.

This effort was performed under AEP Contract C-7693, Release 99-15. Please contact Mr. Don Peck (412-374-2052) or me if you have further questions on this subject.

Donald E Peck /br
W. R. Rice
Customer Projects Manager

Attachment

cc: Rick Kohrt - AEP

SAE-REA-00-559

From: NSBU Systems Analysis Engineering
WIN: 284-5865
Date: March 2, 2000
Subject: Revision to D. C. Cook EQ Dose Report
Ref.: 1. SAE-REA-99-504, "Revised Evaluation of D. C. Cook EQ Dose Based on NUREG-1465 Sources", J. Sejvar, October 27, 1999.

To: D. Peck, 4

cc: S. L. Anderson, 4
U. Bachrach, 4
G. A. Brassart, 4
T. Congedo, STC
J. Monahan, 4

An evaluation of D.C. Cook post-LOCA equipment qualification (EQ) dose, based on the NUREG-1465 source terms, was previously provided in reference 1. At the request of AEP, I have modified the report to improve clarity and to add information relative to the interpretation of the results.

If you have any questions or need additional information, please let me know.

J. Sejvar
Radiation Engineering and Analysis

S. L. Anderson
Radiation Engineering and Analysis

Attachment

D.C. Cook Nuclear Plant
Evaluation of Equipment Qualification Dose
Based on NUREG-1465 Sources

Introduction

The impact of the NUREG-1465[1] source term on post-LOCA equipment qualification (EQ) doses for D.C. Cook Units 1 and 2 was evaluated in support of a submittal to the NRC for acceptance of the use of the NUREG-1465 methodology. The evaluation considered:

1) the release of activity to the containment as a function of time after the accident,
2) the activity concentrations in the containment atmosphere and sump solution, and
3) the dose rates and integrated dose at worst case locations in the containment atmosphere and sump solution.

Parameters and Assumptions

Major parameters and assumptions considered in the analysis include:

Core thermal power	3660 MWt
Cycle length	420 efpd
No. of fuel assemblies	193
Fuel Enrichment	
Region A assemblies (97) assemblies)	3.40 w/o
Region B assemblies (96) assemblies)	3.80 w/o

Fraction of total core fission product inventory released following a Design Basis LOCA:

• Noble gases (Xe, Kr)	1.0
• Halogens (I, Br)	0.40
• Alkali Metals (Cs, Rb)	0.30
• Tellurium Group (Te, Sb, Se)	0.05
• Barium, Strontium (Ba, Sr)	0.02
• Noble Metals (Ru, Rh, Pd, Mo, Tc, Co)	0.0025
• Lanthanides (La, Zr, Nd, Eu, Nb, Pr, Sm, Y, Cm, Am)	0.0002
• Cerium Group (Ce, Pu, Np)	0.0005

[1] NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants - Final Report", USNRC, February 1995.

Fraction of total core fission product inventory for a Severe Accident:
(Additional releases following Design Basis LOCA releases)

	Ex-Vessel	Late-In-Vessel
• Noble gases (Xe, Kr)	0.0	0.0
• Halogens (I, Br)	0.25	0.10
• Alkali Metals (Cs, Rb)	0.35	0.10
• Tellurium Group (Te, Sb, Se)	0.25	0.005
• Barium, Strontium (Ba, Sr)	0.10	-
• Noble Metals (Ru, Rh, Pd, Mo, Tc, Co)	0.0025	-
• Lanthanides (La, Zr, Nd, Eu, Nb, Pr, Sm,Y, Cm, Am)	0.005	-
• Cerium Group (Ce, Pu, Np)	0.005	-

Core release timing	per NUREG-1465
Percent of core activity release from the core in the containment atmosphere after 24 hours	
• Noble gases	100
• Others	0.25
Percent of core activity release from the core in the containment sump water after 24 hours	
• Noble gases	0.0
• Others	100
Containment volume	1.19×10^6 ft^3
Sump water volume	
0 - 2 hours	210,000 gallons
2 - 8 hours	329,000 gallons
8 - 24 hours	353,000 gallons
24 - 48 hours	377,000 gallons
48 - 100 hours	401,000 gallons
100 - 150 hours	431,000 gallons
150 - 200 hours	461,000 gallons
200 - 250 hours	479,000 gallons
250 - 300 hours	491,000 gallons
300 - 336 hours	503,000 gallons
after 336 hours	510,000 gallons

Methodology

Source Term Assumptions

The core activity inventories and releases to the containment following an accident were calculated using the ORIGEN2 computer code.[2] Both a Design Basis Accident (DBA) Loss of Coolant Accident (LOCA) and a Severe Accident were modeled consistent with the release scenarios discussed in NUREG-1465. Although the DBA LOCA is generally limiting for EQ purposes, the Severe Accident sources are included for completeness and for possible use in the evaluation of "equipment survivability" issues.

The removal by decay is considered in the ORIGEN2 runs for various times after the accident. Activity depletion by other removal mechanisms such as containment spray and plate-out are treated in a conservative manner. Even though such removal mechanisms are in effect during the various release phases after the accident (i.e. to 1.97 hours for the design basis LOCA and to 11.97 hours for a severe accident), they are not considered in the analysis until one day after the accident.

For the sources in the containment air, all of the gaseous and not-gaseous activities released from the core are assumed to be dispersed in the containment atmosphere during the first 24 hours of the accident. At this time, it is assumed that 0.25% of the non-gaseous activity released from the core is present in the containment atmosphere. That is, a step removal process is considered which instantaneously transfers all but 0.25% of the halogens and particulates to the sump water.

The sources associated with the sump water sources are conservatively based on the accumulation in the sump water of all of the activity released from the core during the first 24 hours after LOCA. At 24 hours an instantaneous transfer of the gaseous activity to the containment air is assumed. Thus, gaseous activity is conservatively assumed to be retained in the water during the first day after the accident. Further, no credit is taken for retention of airborne iodine and particulates in the containment atmosphere (i.e. the 0.25% of the activity inventory that was assumed to remain in the containment atmosphere as discussed above) or on containment and/or component surfaces. That is, all of the non-gaseous activity that is released from the core is assumed to be retained in the sump water throughout the accident.

The NUREG-1465 release assumptions employed in the analyses are noted to be significantly different from those considered in analyses based on the release assumptions of TID-14844[3]. The release scenario of TID-14844 includes instantaneous release of activity immediately after the accident and release of the following fractions of the core inventory:

Noble gases	1.0
Halogens	0.5
Remaining Fission Product Inventory	0.01

[2] "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials", Allen G. Croff, Nuclear Technology Vol 62, September 1983.

[3] TID-14844, "Calculation of Distance Factors for Power and Test Reactors", U.S. Atomic Energy Commission, J.J. DiNunno, March 1962.

These sources are generally considered to be uniformly distributed in the containment atmosphere throughout the course of the accident. Sources in the sump and water that is recirculated after an accident are generally based on the non-gaseous activity releases (i.e. 50% of the halogens and 1% of the remaining non-gaseous activity) in the sump water. Thus, one of the major differences in the release scenarios between NUREG-1465 and TID-14844 is that the NUREG-1465 sources consider a timed release of activity; whereas the TID-14844 activities are considered to be instantaneously released immediately after the accident. Another major difference is the relatively large amount of cesium activity released from the core based on NUREG-1465 assumptions (i.e. 30% of core inventory for a DBA and 75% for a severe accident) as compared to the TID-14844 fractional release of only 1% of the core inventory. This large amount of cesium becomes an important contributor to long-term dose, owing to the relatively long half-life of the ^{134}Cs and ^{137}Cs isotopes.

Dose Analyses

Gamma dose rates are calculated at the center of the containment atmosphere and in the center of a water-filled sump using a point kernel computer code. No credit for shielding afforded by internal structures or equipment was considered in order to provide conservative estimates of the dose rates received by equipment that is located in air. Dose rates in the sump water are effectively that in an infinite volume of water such that the calculated dose rates within or near the sump solution are maximized. Conservative values of the time-dependent source terms were considered, as described above.

Beta doses were conservatively calculated in the containment air assuming absorption of all of the beta source energy within the beta particle range associated with the average maximum energy of beta particles from fission products; i.e. 1.2 MeV.[4]

The calculated dose rates were integrated over time to obtain integrated dose at these worst-case locations.

[4] Nuclear Reactor Engineering, Glasstone & Sesonske, Van Nostrand Press.

Results

Activity and source term data is included in the attached Tables 1-4 for the DBA and Severe Accident scenarios. Tables 1 and 2 list the sources associated with the containment air and Tables 3 and 4 include the sump water sources.

Dose rates and integrated doses in the containment air and sump water are presented in Figures 1 through 8 for the DBA and Severe Accident scenarios. A summary of the calculated integrated dose at 3 months, 6 months, and 1 year after the accident scenarios is included below:

Time After Accident	Integrated Dose, Rads		
	Containment Atmosphere		Containment Sump
	Gamma	Beta	Gamma
Design Basis LOCA			
3 months	2.7×10^7	4.5×10^8	4.8×10^7
6 months	2.8×10^7	5.0×10^8	7.1×10^7
1 year	2.9×10^7	6.0×10^8	1.2×10^8
Severe Accident			
3 months	4.6×10^7	5.8×10^8	1.7×10^8
6 months	4.8×10^7	6.34×10^8	2.5×10^8
1 year	5.0×10^7	7.5×10^8	4.1×10^8

Note that the dose rates within the sump water exceed those that are due to the activity in the containment atmosphere. It should also be noted that since the dose rates at the surface of the sump water would be approximately ½ of the value listed above for the containment sump, the proximity of equipment to the sump water is an important consideration in evaluating equipment doses inside of the containment.

The integrated gamma and beta dose rates in the containment atmosphere following a design basis LOCA and a severe accident are summarized in Figures 9 and 10. These figures also include a comparison of the calculated doses to "bounding" values reported in the Westinghouse Equipment Qualification Methodology report, WCAP-8587.[5] The figures indicate that the calculated integrated gamma doses in the containment air after one year are factors of approximately 4 and 2 ½ lower than the WCAP-8587 dose for the Design Basis LOCA and the Severe Accident scenarios, respectively. The integrated beta doses after one year are about a factor of 2 lower than the WCAP-8587 values.

The integrated gamma doses in the sump water for the NUREG-1465 releases are depicted in Figure 11. A comparison of these results to the NUREG-1465 doses calculated in the containment air (Figure 9) indicates that the containment air gamma doses exceed the sump water gamma doses for the first 100 hours after an accident.

[5] WCAP-8587, Revision 6-A (NP), "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment", G. Butterworth and A. Ball, March 1983.

After about 100 hours, the integrated doses in the sump continue to increase whereas the doses in the air remain relatively constant. The integrated gamma dose in the sump water at one year after the DBA is approximately 4 times the integrated dose that is calculated in the containment atmosphere. Similarly, the sump dose for the Severe Accident after one year is about 8 times higher than the Severe Accident dose in the containment air.

The difference in the long-term dose trends in the sump as compared to the containment air is related to the fact that the NUREG-1465 source terms associated with the sump include a much larger long-lived cesium component. The sump source is based on 30% and 75% of the core cesium inventory for the design basis and severe accident scenarios, respectively; whereas the containment air includes only 0.25% of these activities after 24 hours.

Figure 11 also includes a comparison of the integrated DBA and Severe Accident sump doses to the doses at the center of a pipe that contains recirculating sump water fluid. The piping doses are those presented in the Westinghouse Equipment Qualification Methodology report, WCAP-8587. It is important to note that the WCAP values are based on a different source geometry than that considered for the NUREG-1465 sump water dose analysis; the NUREG-1465 geometry assumption is more conservative than that considered in the WCAP. However, even with the more conservative geometry the doses associated with NUREG-1465 DBA source terms are lower than that presented in WCAP-8587 through one year after the accident.

Further, the Severe Accident doses are lower than the WCAP values for about the first 40 days after the accident. After this time, the integrated doses associated with the NUREG-1465 source term assumptions increase relatively rapidly, owing to the large (long-lived) cesium inventory in the sump water.

Thus, even with the more conservative source geometry assumptions, the sump water doses for the NUREG-1465 DBA sources are lower than WCAP-8587 doses. For the Severe Accident, the values are lower than the WCAP doses for the first 40 days after the accident and higher than the WCAP values thereafter. As mentioned above in the methodology section, the DBA values are generally limiting for EQ purposes and the Severe Accident sources are included for completeness and for possible use in the evaluation of "equipment survivability" issues.

TABLE 1
DBA ACCIDENT CONTAINMENT AIR SHIELDING SOURCE STRENGTHS - MeV/watt-sec
Gap-init. release (inst.) at 10 min / Late Gap (10-40 min) / Early in-vessel release (40 min - 1.97 hr)
/ No additional releases after 1.97 hours for Design Basis Accident / Releases to 11.97 hours for Severe Accident

Energy (Mev)	Time after Accident											
	Shutdown	30 sec	9.99 min	10 min	40 min	59.5 min	79 min	1.97 hr	2.97 hr	3.97 hr	5.97 hr	11.97 hr
< 0.15	0.00E+00	0.00E+00	0.00E+00	1.57E+07	1.78E+07	1.12E+08	1.59E+08	2.01E+08	1.81E+08	1.66E+08	1.48E+08	1.25E+08
0.15 - 0.45	0.00E+00	0.00E+00	0.00E+00	4.54E+07	5.38E+07	2.89E+08	4.05E+08	5.34E+08	4.71E+08	4.28E+08	3.73E+08	2.90E+08
0.45 - 1.0	0.00E+00	0.00E+00	0.00E+00	2.98E+08	3.89E+08	1.22E+09	1.77E+09	2.46E+09	1.80E+09	1.45E+09	1.07E+09	6.79E+08
1.0 - 1.5	0.00E+00	0.00E+00	0.00E+00	2.06E+08	2.40E+08	6.46E+08	8.83E+08	1.12E+09	8.53E+08	7.04E+08	5.33E+08	2.88E+08
1.5 - 2.0	0.00E+00	0.00E+00	0.00E+00	7.33E+07	8.72E+07	3.62E+08	5.16E+08	6.57E+08	5.36E+08	4.37E+08	3.06E+08	1.32E+08
2.0 - 2.5	0.00E+00	0.00E+00	0.00E+00	7.88E+07	8.21E+07	5.47E+08	6.97E+08	7.37E+08	5.59E+08	4.34E+08	2.70E+08	7.11E+07
2.5 - 3.0	0.00E+00	0.00E+00	0.00E+00	2.69E+07	2.48E+07	1.08E+08	1.32E+08	1.22E+08	7.52E+07	4.88E+07	2.34E+07	4.55E+06
3.0 - 4.0	0.00E+00	0.00E+00	0.00E+00	9.61E+06	5.19E+06	1.56E+07	1.94E+07	1.87E+07	1.24E+07	8.74E+06	4.87E+06	1.05E+06
4.0 - 6.0	0.00E+00	0.00E+00	0.00E+00	2.94E+06	3.81E+05	1.84E+06	2.86E+06	3.50E+06	3.04E+06	2.40E+06	1.47E+06	3.39E+05
6.0 -11.0	0.00E+00	0.00E+00	0.00E+00	6.06E-03	3.00E-09	2.81E-05	5.61E-05	1.12E-04	1.12E-04	1.12E-04	1.12E-04	1.12E-04
Total G-Mev/watt-s	0.00E+00	0.00E+00	0.00E+00	7.57E+08	9.00E+08	3.30E+09	4.59E+09	5.85E+09	4.49E+09	3.68E+09	2.73E+09	1.59E+09
G Power (watts)	0.00E+00	0.00E+00	0.00E+00	4.35E+05	5.17E+05	1.90E+06	2.64E+06	3.37E+06	2.59E+06	2.11E+06	1.57E+06	9.15E+05
Thermal (watts)	0.00E+00	0.00E+00	0.00E+00	6.35E+05	7.26E+05	2.84E+06	3.97E+06	5.03E+06	3.97E+06	3.29E+06	2.48E+06	1.50E+06
Beta - Mev/watt-s	0.00E+00	0.00E+00	0.00E+00	3.48E+08	3.63E+08	1.64E+09	2.32E+09	2.90E+09	2.41E+09	2.04E+09	1.58E+09	1.02E+09

Energy (Mev)	Time after Shutdown										
	No additional releases										
	24hr	24+hr	64hr	72hr	100h	7dy	30d	90d	180d	1yr	5yr
< 0.15	1.09E+08	9.49E+07	7.43E+07	7.11E+07	6.10E+07	4.21E+07	2.10E+06	5.66E+04	5.36E+04	5.18E+04	3.98E+04
0.15 - 0.45	2.08E+08	6.44E+07	4.33E+06	2.91E+06	1.23E+06	6.20E+05	6.20E+04	1.83E+04	1.73E+04	1.67E+04	1.27E+04
0.45 - 1.0	4.70E+08	1.07E+07	9.07E+05	7.50E+05	5.25E+05	3.92E+05	2.31E+05	1.98E+05	1.84E+05	1.63E+05	7.56E+04
1.0 - 1.5	1.16E+08	5.02E+05	8.76E+04	7.97E+04	6.28E+04	4.42E+04	1.45E+04	7.21E+03	6.35E+03	5.35E+03	1.39E+03
1.5 - 2.0	4.69E+07	5.85E+05	8.63E+04	8.96E+04	9.73E+04	9.76E+04	3.01E+04	1.23E+03	4.51E+01	1.64E+01	1.58E+00
2.0 - 2.5	9.27E+06	2.88E+06	5.70E+03	5.22E+03	4.36E+03	2.98E+03	4.70E+02	5.10E+01	2.54E+01	1.55E+01	6.08E-01
2.5 - 3.0	1.28E+06	4.69E+04	4.97E+03	5.21E+03	5.72E+03	5.78E+03	1.80E+03	7.10E+01	1.81E+00	9.03E-01	5.73E-02
3.0 - 4.0	6.57E+04	1.83E+02	5.41E+01	5.40E+01	5.93E+01	6.03E+01	1.91E+01	9.74E-01	2.15E-01	1.48E-01	9.48E-03
4.0 - 6.0	1.80E+04	4.49E+01	1.03E+00	1.46E-01	1.58E-04	1.49E-06	1.43E-06	1.29E-06	1.14E-06	9.65E-07	7.12E-07
6.0 -11.0	1.12E-04	2.81E-07	2.80E-07	2.80E-07	2.79E-07	2.78E-07	2.66E-07	2.41E-07	2.13E-07	1.80E-07	1.33E-07
Total G - Mev/watt-s	9.60E+08	1.74E+08	7.98E+07	7.50E+07	6.29E+07	4.32E+07	2.44E+06	2.81E+05	2.62E+05	2.37E+05	1.29E+05
G Power (watts)	5.52E+05	1.00E+05	4.59E+04	4.31E+04	3.62E+04	2.49E+04	1.41E+03	1.62E+02	1.51E+02	1.36E+02	7.45E+01
Thermal (watts)	9.46E+05	3.06E+05	1.74E+05	1.65E+05	1.39E+05	9.59E+04	6.11E+03	1.38E+03	1.34E+03	1.28E+03	9.55E+02
Beta - Mev/watt-s	6.85E+08	3.57E+08	2.22E+08	2.11E+08	1.79E+08	1.24E+08	8.18E+06	2.11E+06	2.06E+06	1.99E+06	1.53E+06

TABLE 2
SEVERE ACCIDENT CONTAINMENT AIR SHIELDING SOURCE STRENGTHS - MeV/watt-sec
Gap-init. release (inst.) at 10 min / Late Gap (10-40 min) / Early in-vessel release (40 min - 1.97 hr)
/ No additional releases after 1.97 hours for Design Basis Accident / Releases to 11.97 hours for Severe Accident

Energy (Mev)	Time after Accident											
	Shutdown	30 sec	9.99 min	10 min	40 min	59.5 min	79 min	1.97 hr	2.97 hr	3.97 hr	5.97 hr	11.97 hr
< 0.15	0.00E+00	0.00E+00	0.00E+00	1.57E+07	1.78E+07	1.12E+08	1.59E+08	2.01E+08	2.15E+08	2.23E+08	2.10E+08	1.84E+08
0.15 - 0.45	0.00E+00	0.00E+00	0.00E+00	4.54E+07	5.38E+07	2.89E+08	4.05E+08	5.34E+08	6.12E+08	6.69E+08	6.66E+08	6.15E+08
0.45 - 1.0	0.00E+00	0.00E+00	0.00E+00	2.98E+08	3.89E+08	1.22E+09	1.77E+09	2.46E+09	2.81E+09	3.06E+09	2.60E+09	2.09E+09
1.0 - 1.5	0.00E+00	0.00E+00	0.00E+00	2.06E+08	2.40E+08	6.46E+08	8.83E+08	1.12E+09	1.27E+09	1.34E+09	1.09E+09	6.93E+08
1.5 - 2.0	0.00E+00	0.00E+00	0.00E+00	7.33E+07	8.72E+07	3.62E+08	5.16E+08	6.57E+08	6.75E+08	6.50E+08	5.03E+08	2.90E+08
2.0 - 2.5	0.00E+00	0.00E+00	0.00E+00	7.88E+07	8.21E+07	5.47E+08	6.97E+08	7.37E+08	5.91E+08	4.80E+08	3.20E+08	1.04E+08
2.5 - 3.0	0.00E+00	0.00E+00	0.00E+00	2.69E+07	2.48E+07	1.08E+08	1.32E+08	1.21E+08	8.01E+07	5.32E+07	2.72E+07	8.22E+06
3.0 - 4.0	0.00E+00	0.00E+00	0.00E+00	9.61E+06	5.19E+06	1.56E+07	1.94E+07	1.87E+07	1.34E+07	9.49E+06	5.31E+06	1.15E+06
4.0 - 6.0	0.00E+00	0.00E+00	0.00E+00	2.94E+06	3.81E+05	1.84E+06	2.86E+06	3.50E+06	3.08E+06	2.42E+06	1.54E+06	3.55E+05
6.0 -11.0	0.00E+00	0.00E+00	0.00E+00	6.06E-03	3.00E-09	2.81E-05	5.61E-05	1.12E-04	1.50E-03	2.89E-03	2.89E-03	2.89E-03
Total G-Mev/watt-s	0.00E+00	0.00E+00	0.00E+00	7.57E+08	9.00E+08	3.30E+09	4.59E+09	5.85E+09	6.26E+09	6.49E+09	5.42E+09	3.99E+09
G Power (watts)	0.00E+00	0.00E+00	0.00E+00	4.35E+05	5.17E+05	1.90E+06	2.64E+06	3.37E+06	3.60E+06	3.73E+06	3.12E+06	2.29E+06
Thermal (watts)	0.00E+00	0.00E+00	0.00E+00	6.35E+05	7.26E+05	2.84E+06	3.97E+06	5.03E+06	5.34E+06	5.48E+06	4.64E+06	3.44E+06
Beta - Mev/watt-s	0.00E+00	0.00E+00	0.00E+00	3.48E+08	3.63E+08	1.64E+09	2.32E+09	2.90E+09	3.03E+09	3.05E+09	2.64E+09	1.99E+09

Energy (Mev)	Time after Shutdown										
	No additional releases										
	24hr	24+hr	64hr	72hr	100h	7dy	30d	90d	180d	1yr	5yr
< 0.15	1.58E+08	1.05E+08	8.12E+07	7.77E+07	6.66E+07	4.59E+07	2.30E+06	6.41E+04	5.93E+04	5.65E+04	4.30E+04
0.15 - 0.45	4.72E+08	1.13E+08	7.16E+06	4.68E+06	1.84E+06	9.41E+05	1.07E+05	2.33E+04	2.07E+04	1.94E+04	1.42E+04
0.45 - 1.0	1.54E+09	2.13E+07	2.89E+06	2.55E+06	1.97E+06	1.43E+06	6.50E+05	5.11E+05	4.65E+05	4.04E+05	1.78E+05
1.0 - 1.5	3.24E+08	1.04E+06	3.31E+05	3.05E+05	2.43E+05	1.60E+05	3.93E+04	1.93E+04	1.69E+04	1.43E+04	3.74E+03
1.5 - 2.0	1.73E+08	9.23E+05	5.16E+05	5.38E+05	5.85E+05	5.85E+05	1.80E+05	7.30E+03	2.11E+02	5.55E+01	8.01E+00
2.0 - 2.5	2.73E+07	3.06E+06	3.38E+04	3.19E+04	2.70E+04	1.85E+04	3.04E+03	3.81E+02	2.04E+02	1.23E+02	3.81E+00
2.5 - 3.0	6.91E+06	6.30E+04	2.99E+04	3.14E+04	3.44E+04	3.47E+04	1.08E+04	4.19E+02	5.88E+00	1.91E+00	1.18E-01
3.0 - 4.0	1.24E+05	3.30E+02	3.09E+02	3.22E+02	3.55E+02	3.60E+02	1.13E+02	4.85E+00	4.51E-01	2.97E-01	1.90E-02
4.0 - 6.0	1.88E+04	4.71E+01	1.08E+00	1.53E-01	2.03E-04	3.83E-05	3.67E-05	3.32E-05	2.93E-05	2.47E-05	1.81E-05
6.0 -11.0	2.89E-03	7.21E-06	7.20E-06	7.19E-06	7.18E-06	7.14E-06	6.85E-06	6.19E-06	5.47E-06	4.60E-06	3.38E-06
Total G - Mev/watt-s	2.70E+09	2.44E+08	9.22E+07	8.58E+07	7.13E+07	4.91E+07	3.29E+06	6.25E+05	5.62E+05	4.94E+05	2.39E+05
G Power (watts)	1.55E+06	1.41E+05	5.30E+04	4.94E+04	4.10E+04	2.82E+04	1.89E+03	3.60E+02	3.23E+02	2.84E+02	1.37E+02
Thermal (watts)	2.35E+06	3.98E+05	1.95E+05	1.83E+05	1.54E+05	1.06E+05	7.14E+03	1.71E+03	1.62E+03	1.53E+03	1.09E+03
Beta - Mev/watt-s	1.39E+09	4.48E+08	2.46E+08	2.33E+08	1.97E+08	1.35E+08	9.12E+06	2.35E+06	2.25E+06	2.16E+06	1.65E+06

TABLE 3
DBA ACCIDENT SUMP WATER SHIELDING SOURCE STRENGTHS - MeV/watt-sec
Gap-init. release (inst.) at 10 min / Late Gap (10-40 min) / Early in-vessel release (40 min - 1.97 hr)
/ No additional releases after 1.97 hours for Design Basis Accident / Releases to 11.97 hours for Severe Accident

Energy (Mev)	Time after Accident											
	Shutdown	30s rel.	9.9m	10m tot.	40m rel.	59.5m rel	79m rel.	1.97hr	2.97hr	3.97hr	5.97hr	11.97h
< 0.15	0.00E+00	0.00E+00	0.00E+00	1.57E+07	1.78E+07	1.12E+08	1.59E+08	2.01E+08	1.81E+08	1.66E+08	9.44E+07	7.89E+07
0.15 - 0.45	0.00E+00	0.00E+00	0.00E+00	4.54E+07	5.38E+07	2.89E+08	4.05E+08	5.34E+08	4.71E+08	4.28E+08	2.38E+08	1.84E+08
0.45 - 1.0	0.00E+00	0.00E+00	0.00E+00	2.98E+08	3.89E+08	1.22E+09	1.77E+09	2.46E+09	1.80E+09	1.45E+09	6.81E+08	4.30E+08
1.0 - 1.5	0.00E+00	0.00E+00	0.00E+00	2.06E+08	2.40E+08	6.46E+08	8.83E+08	1.12E+09	8.52E+08	7.04E+08	3.40E+08	1.82E+08
1.5 - 2.0	0.00E+00	0.00E+00	0.00E+00	7.33E+07	8.72E+07	3.62E+08	5.16E+08	6.57E+08	5.36E+08	4.37E+08	1.95E+08	8.32E+07
2.0 - 2.5	0.00E+00	0.00E+00	0.00E+00	7.88E+07	8.21E+07	5.47E+08	6.97E+08	7.37E+08	5.59E+08	4.34E+08	1.72E+08	4.50E+07
2.5 - 3.0	0.00E+00	0.00E+00	0.00E+00	2.69E+07	2.48E+07	1.08E+08	1.32E+08	1.21E+08	7.52E+07	4.88E+07	1.49E+07	2.88E+06
3.0 - 4.0	0.00E+00	0.00E+00	0.00E+00	9.61E+06	5.19E+06	1.56E+07	1.94E+07	1.87E+07	1.24E+07	8.74E+06	3.11E+06	6.67E+05
4.0 - 6.0	0.00E+00	0.00E+00	0.00E+00	2.94E+06	3.81E+05	1.84E+06	2.86E+06	3.50E+06	3.04E+06	2.40E+06	9.37E+05	2.15E+05
6.0 -11.0	0.00E+00	0.00E+00	0.00E+00	6.06E-03	3.00E-09	2.81E-05	5.61E-05	1.12E-04	1.12E-04	1.12E-04	7.17E-05	7.10E-05
Total G-Mev/watt-s	0.00E+00	0.00E+00	0.00E+00	7.57E+08	9.00E+08	3.30E+09	4.59E+09	5.85E+09	4.49E+09	3.68E+09	1.74E+09	1.01E+09
G Power (watts)	0.00E+00	0.00E+00	0.00E+00	4.35E+05	5.17E+05	1.90E+06	2.64E+06	3.37E+06	2.59E+06	2.11E+06	1.00E+06	5.79E+05
Thermal (watts)	0.00E+00	0.00E+00	0.00E+00	6.35E+05	7.26E+05	2.84E+06	3.97E+06	5.03E+06	3.97E+06	3.29E+06	1.58E+06	9.48E+05
Beta - Mev/watt-s	0.00E+00	0.00E+00	0.00E+00	3.48E+08	3.63E+08	1.64E+09	2.32E+09	2.90E+09	2.41E+09	2.04E+09	1.01E+09	6.43E+08

Energy (Mev)	Time after Shutdown										
	No additional releases										
	24hr	24+hr	64hr	72hr	100h	7dy	30d	90d	180d	1yr	5yr
< 0.15	6.77E+07	8.77E+06	6.01E+06	5.65E+06	4.65E+06	3.01E+06	6.83E+05	2.42E+05	1.87E+05	1.55E+05	9.17E+04
0.15 - 0.45	1.29E+08	8.95E+07	6.97E+07	6.63E+07	5.64E+07	3.85E+07	5.66E+06	1.86E+05	1.06E+05	8.31E+04	3.65E+04
0.45 - 1.0	2.92E+08	2.86E+08	1.51E+08	1.38E+08	1.08E+08	7.26E+07	4.14E+07	2.94E+07	2.72E+07	2.38E+07	1.01E+07
1.0 - 1.5	7.22E+07	7.21E+07	2.03E+07	1.83E+07	1.38E+07	8.61E+06	2.83E+06	1.19E+06	1.05E+06	8.81E+05	2.29E+05
1.5 - 2.0	2.92E+07	2.89E+07	2.02E+07	2.07E+07	2.14E+07	1.90E+07	5.87E+06	2.03E+05	7.42E+03	2.70E+03	2.60E+02
2.0 - 2.5	5.76E+06	3.98E+06	1.29E+06	1.20E+06	9.60E+05	5.80E+05	9.16E+04	8.39E+03	4.19E+03	2.55E+03	1.00E+02
2.5 - 3.0	7.99E+05	7.71E+05	1.16E+06	1.20E+06	1.26E+06	1.13E+06	3.50E+05	1.17E+04	2.97E+02	1.49E+02	9.44E+00
3.0 - 4.0	4.08E+04	4.08E+04	1.19E+04	1.24E+04	1.31E+04	1.17E+04	3.71E+03	1.61E+02	3.54E+01	2.44E+01	1.56E+00
4.0 - 6.0	1.12E+04	1.12E+04	3.51E-04	3.47E-04	3.30E-04	2.91E-04	2.79E-04	2.13E-04	1.89E-04	1.59E-04	1.17E-04
6.0 -11.0	6.98E-05	6.98E-05	6.54E-05	6.46E-05	6.15E-05	5.41E-05	5.19E-05	3.97E-05	3.52E-05	2.96E-05	2.19E-05
Total G - Mev/watt-s	5.97E+08	4.90E+08	2.70E+08	2.52E+08	2.06E+08	1.43E+08	5.68E+07	3.13E+07	2.86E+07	2.50E+07	1.05E+07
G Power (watts)	3.43E+05	2.82E+05	1.55E+05	1.45E+05	1.19E+05	8.25E+04	3.27E+04	1.80E+04	1.65E+04	1.44E+04	6.04E+03
Thermal (watts)	5.88E+05	3.99E+05	2.17E+05	2.02E+05	1.64E+05	1.13E+05	4.34E+04	2.26E+04	2.02E+04	1.76E+04	8.01E+03
Beta - Mev/watt-s	4.26E+08	2.04E+08	1.08E+08	9.93E+07	7.87E+07	5.34E+07	1.86E+07	8.08E+06	6.55E+06	5.58E+06	3.43E+06

TABLE 4
SEVERE ACCIDENT SUMP SHIELDING SOURCE STRENGTHS - MeV/watt-sec
Gap-init. release (inst.) at 10 min / Late Gap (10-40 min) / Early In-vessel release (40 min - 1.97 hr)
/ No additional releases after 1.97 hours for Design Basis Accident / Releases to 11.97 hours for Severe Accident

Energy (Mev)	Time after Accident											
	Shutdown	30 sec	9.99 min	10 min	40 min	59.5 min	79 min	1.97 hr	2.97 hr	3.97 hr	5.97 hr	11.97 hr
< 0.15	0.00E+00	0.00E+00	0.00E+00	1.57E+07	1.78E+07	1.12E+08	1.59E+08	2.01E+08	2.15E+08	2.23E+08	1.34E+08	1.17E+08
0.15 - 0.45	0.00E+00	0.00E+00	0.00E+00	4.54E+07	5.38E+07	2.89E+08	4.05E+08	5.34E+08	6.12E+08	6.69E+08	4.25E+08	3.89E+08
0.45 - 1.0	0.00E+00	0.00E+00	0.00E+00	2.98E+08	3.89E+08	1.22E+09	1.77E+09	2.46E+09	2.81E+09	3.06E+09	1.66E+09	1.32E+09
1.0 - 1.5	0.00E+00	0.00E+00	0.00E+00	2.06E+08	2.40E+08	6.46E+08	8.83E+08	1.12E+09	1.27E+09	1.34E+09	6.98E+08	4.39E+08
1.5 - 2.0	0.00E+00	0.00E+00	0.00E+00	7.33E+07	8.72E+07	3.62E+08	5.16E+08	6.57E+08	6.75E+08	6.50E+08	3.21E+08	1.83E+08
2.0 - 2.5	0.00E+00	0.00E+00	0.00E+00	7.88E+07	8.21E+07	5.47E+08	6.97E+08	7.37E+08	5.91E+08	4.80E+08	2.04E+08	6.58E+07
2.5 - 3.0	0.00E+00	0.00E+00	0.00E+00	2.69E+07	2.48E+07	1.08E+08	1.32E+08	1.21E+08	8.01E+07	5.32E+07	1.74E+07	5.20E+06
3.0 - 4.0	0.00E+00	0.00E+00	0.00E+00	9.61E+06	5.19E+06	1.56E+07	1.94E+07	1.87E+07	1.34E+07	9.49E+06	3.39E+06	7.27E+05
4.0 - 6.0	0.00E+00	0.00E+00	0.00E+00	2.94E+06	3.81E+05	1.84E+06	2.86E+06	3.50E+06	3.08E+06	2.42E+06	9.82E+05	2.25E+05
6.0 -11.0	0.00E+00	0.00E+00	0.00E+00	6.06E-03	3.00E-09	2.81E-05	5.61E-05	1.12E-04	1.50E-03	2.89E-03	1.84E-03	1.83E-03
Total G-Mev/watt-s	0.00E+00	0.00E+00	0.00E+00	7.57E+08	9.00E+08	3.30E+09	4.59E+09	5.85E+09	6.26E+09	6.49E+09	3.46E+09	2.52E+09
G Power (watts)	0.00E+00	0.00E+00	0.00E+00	4.35E+05	5.17E+05	1.90E+06	2.64E+06	3.37E+06	3.60E+06	3.73E+06	1.99E+06	1.45E+06
Thermal (watts)	0.00E+00	0.00E+00	0.00E+00	6.35E+05	7.26E+05	2.84E+06	3.97E+06	5.03E+06	5.34E+06	5.48E+06	2.96E+06	2.17E+06
Beta - Mev/watt-s	0.00E+00	0.00E+00	0.00E+00	3.48E+08	3.63E+08	1.64E+09	2.32E+09	2.90E+09	3.03E+09	3.05E+09	1.68E+09	1.26E+09

Energy (Mev)	Time after Shutdown										
	No additional releases										
	24hr	24+hr	64hr	72hr	100h	7dy	30d	90d	180d	1yr	5yr
< 0.15	9.84E+07	3.30E+07	2.29E+07	2.15E+07	1.75E+07	1.11E+07	2.95E+06	1.03E+06	7.05E+05	5.39E+05	3.06E+05
0.15 - 0.45	2.93E+08	2.24E+08	1.72E+08	1.63E+08	1.37E+08	9.15E+07	1.40E+07	8.69E+05	5.20E+05	3.99E+05	1.79E+05
0.45 - 1.0	9.56E+08	9.45E+08	5.71E+08	5.30E+08	4.22E+08	2.74E+08	1.23E+08	8.08E+07	7.33E+07	6.34E+07	2.69E+07
1.0 - 1.5	2.01E+08	2.01E+08	7.71E+07	7.03E+07	5.34E+07	3.11E+07	7.66E+06	3.18E+06	2.79E+06	2.35E+06	6.15E+05
1.5 - 2.0	1.08E+08	1.07E+08	1.21E+08	1.24E+08	1.29E+08	1.14E+08	3.52E+07	1.20E+06	3.47E+04	9.14E+03	1.32E+03
2.0 - 2.5	1.70E+07	1.51E+07	7.86E+06	7.36E+06	5.94E+06	3.61E+06	5.93E+05	6.27E+04	3.36E+04	2.03E+04	6.27E+02
2.5 - 3.0	4.30E+06	4.27E+06	6.99E+06	7.23E+06	7.57E+06	6.75E+06	2.10E+06	6.90E+04	9.69E+02	3.14E+02	1.94E+01
3.0 - 4.0	7.71E+04	7.71E+04	7.15E+04	7.42E+04	7.82E+04	7.02E+04	2.20E+04	7.98E+02	7.43E+01	4.89E+01	3.13E+00
4.0 - 6.0	1.17E+04	1.17E+04	9.03E-03	8.90E-03	8.49E-03	7.47E-03	7.16E-03	5.47E-03	4.83E-03	4.06E-03	2.98E-03
6.0 -11.0	1.79E-03	1.79E-03	1.68E-03	1.66E-03	1.58E-03	1.39E-03	1.33E-03	1.02E-03	9.01E-04	7.58E-04	5.56E-04
Total G - Mev/watt-s	1.68E+09	1.53E+09	9.79E+08	9.24E+08	7.73E+08	5.32E+08	1.85E+08	8.72E+07	7.74E+07	6.67E+07	2.80E+07
G Power (watts)	9.65E+05	8.79E+05	5.63E+05	5.31E+05	4.44E+05	3.06E+05	1.07E+05	5.02E+04	4.45E+04	3.84E+04	1.61E+04
Thermal (watts)	1.46E+06	1.22E+06	7.72E+05	7.28E+05	6.08E+05	4.22E+05	1.51E+05	6.78E+04	5.72E+04	4.85E+04	2.23E+04
Beta - Mev/watt-s	8.65E+08	5.90E+08	3.64E+08	3.42E+08	2.84E+08	2.01E+08	7.70E+07	3.07E+07	2.21E+07	1.76E+07	1.08E+07

Figure 1
Gamma Dose in Containment After Design Basis Accident

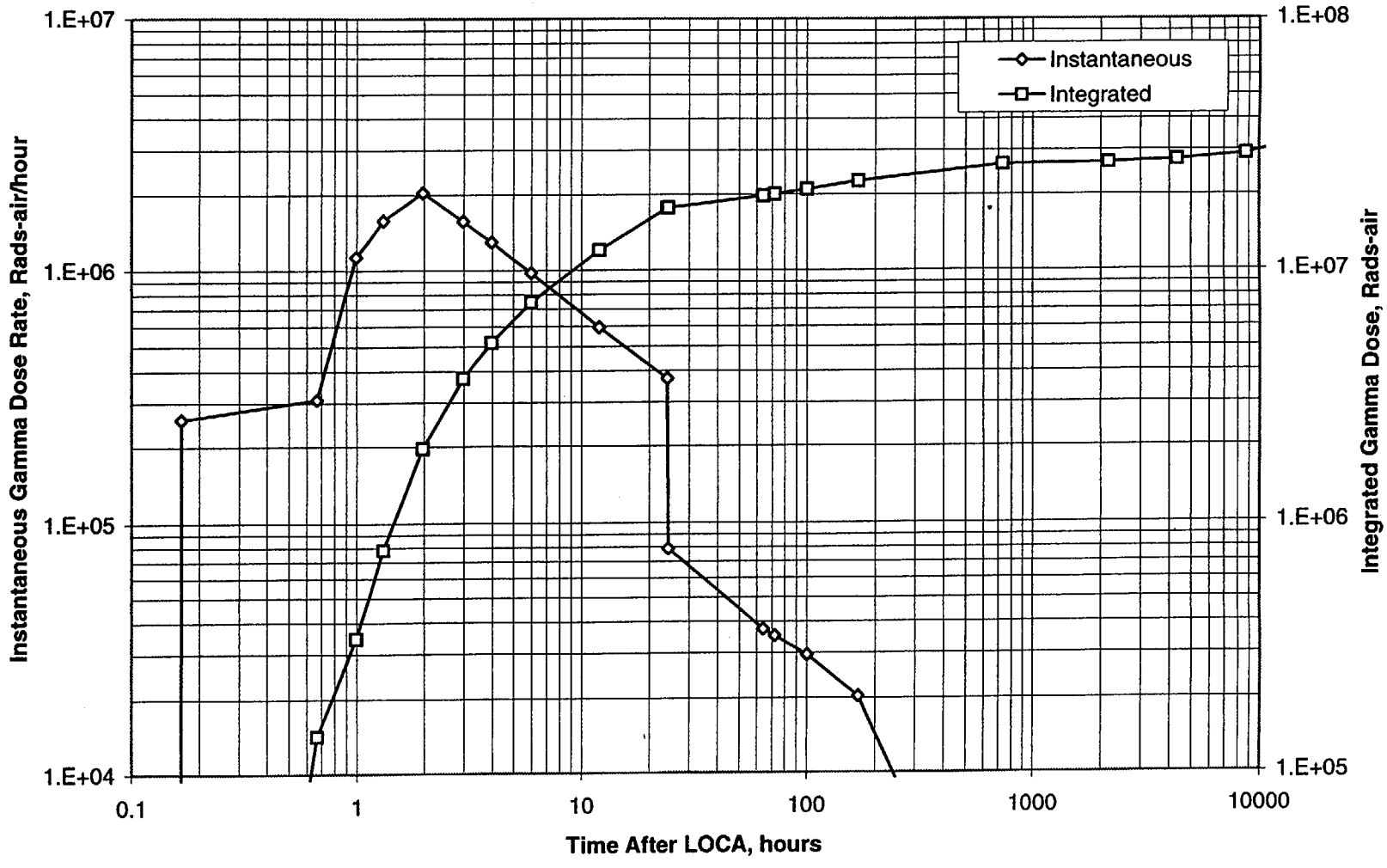


Figure 2
Beta Dose in Containment After Design Basis Accident

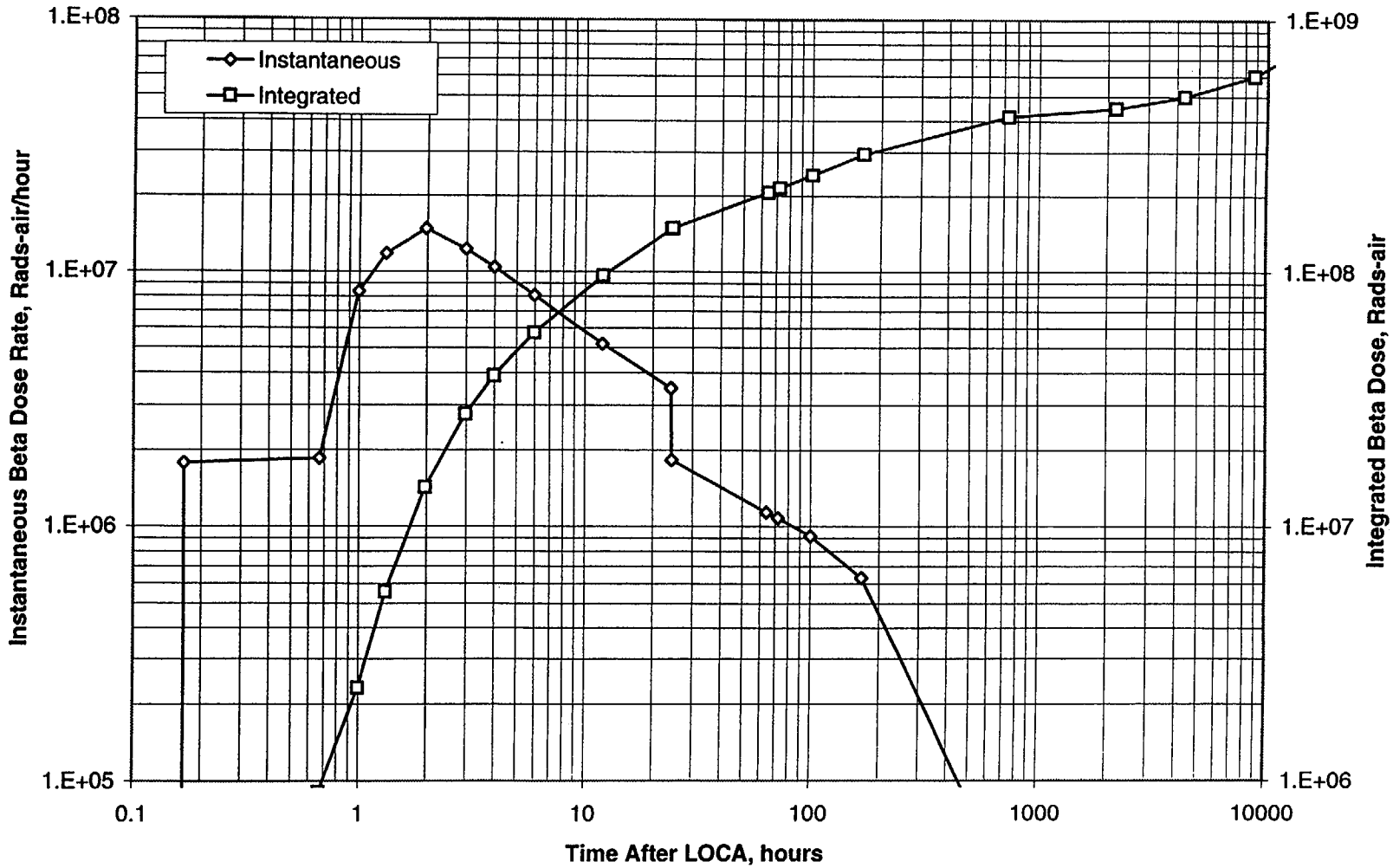


Figure 3
Gamma Dose in Containment After Severe Accident

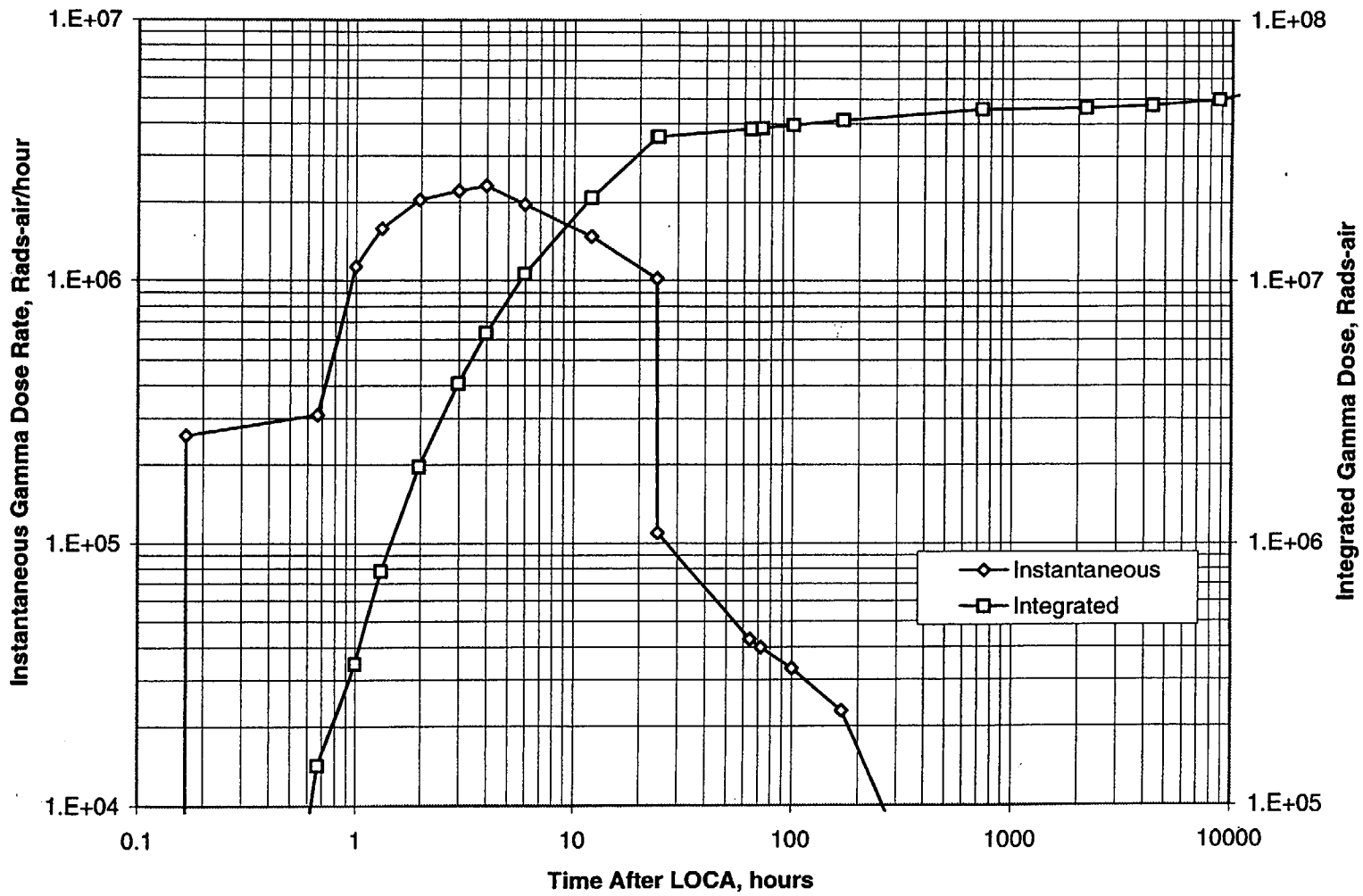


Figure 4
Beta Dose in Containment After Severe Accident

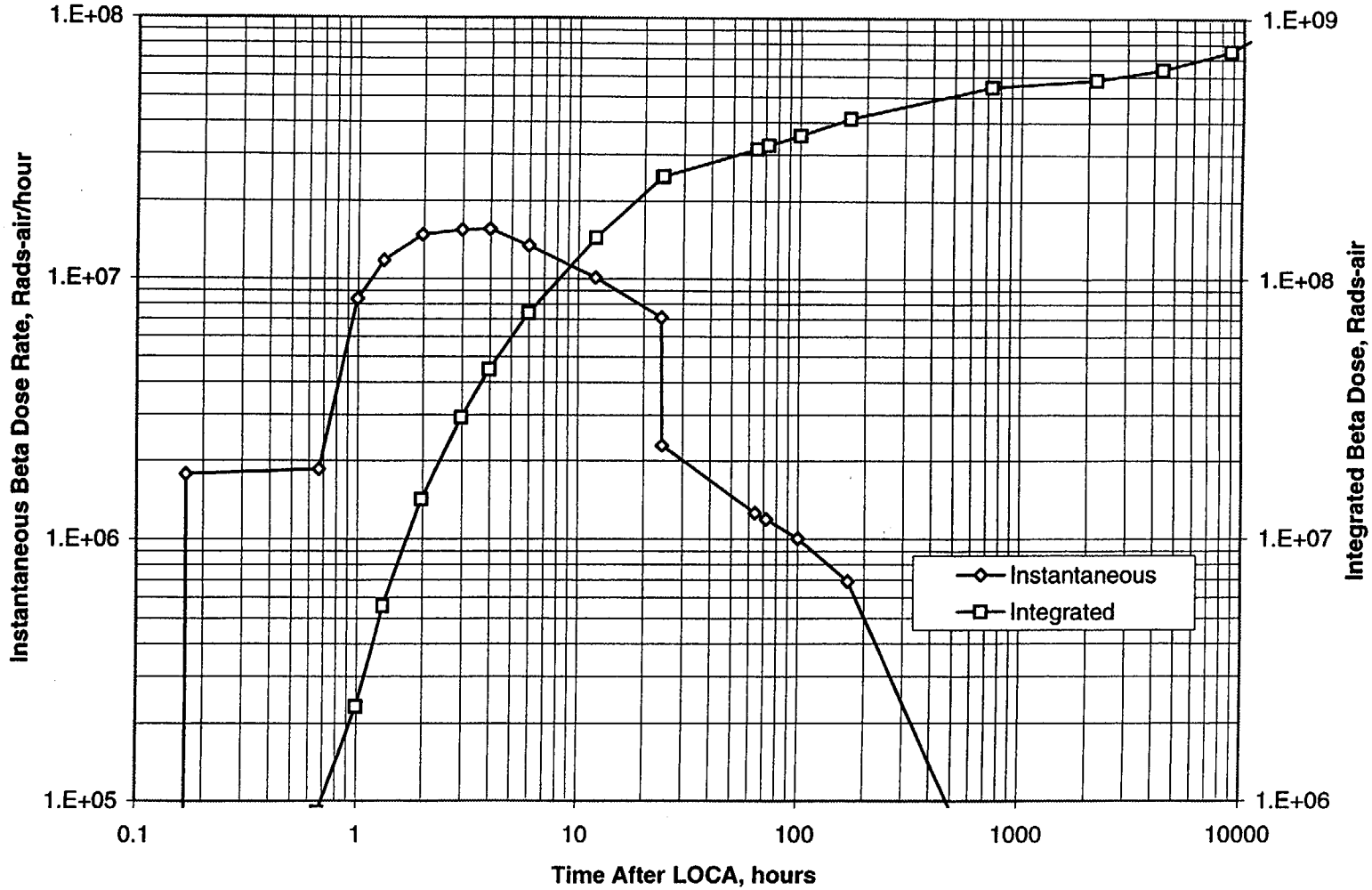


Figure 5
Gamma Dose in Containment Sump Water After Design Basis Accident

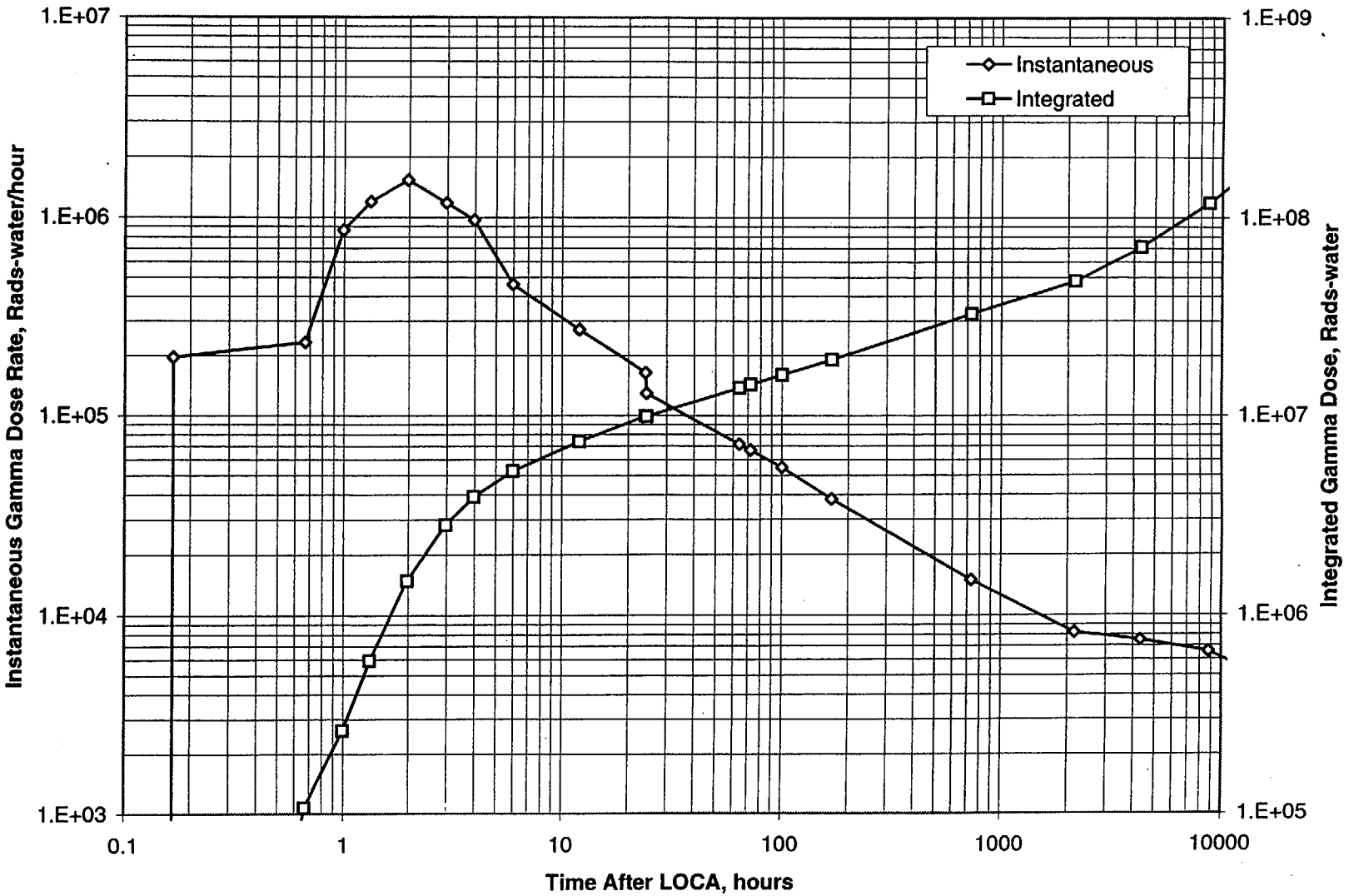


Figure 6
Beta Dose in Containment Sump Water After Design Basis Accident

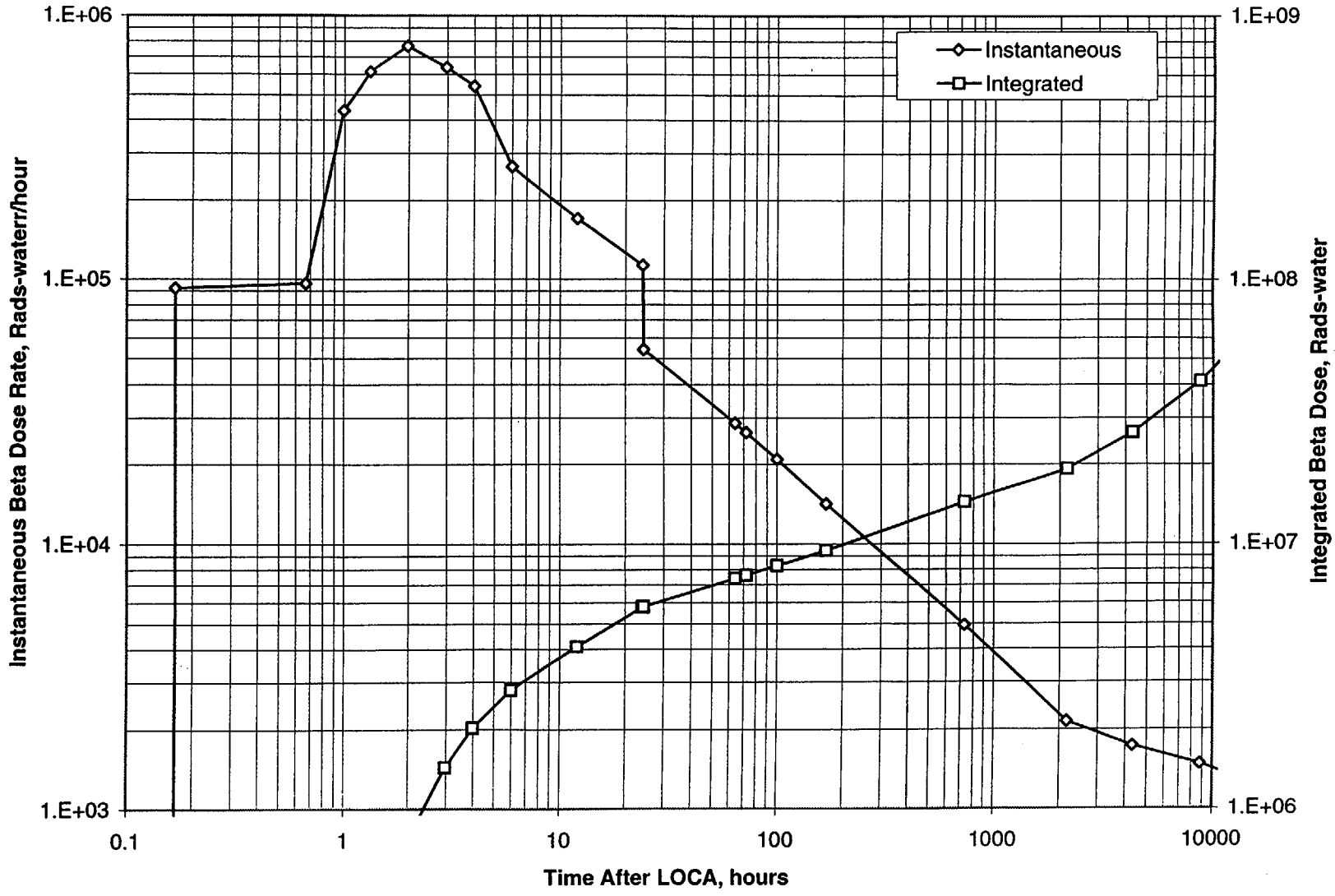


Figure 7
Gamma Dose in Containment Sump Water After Severe Accident

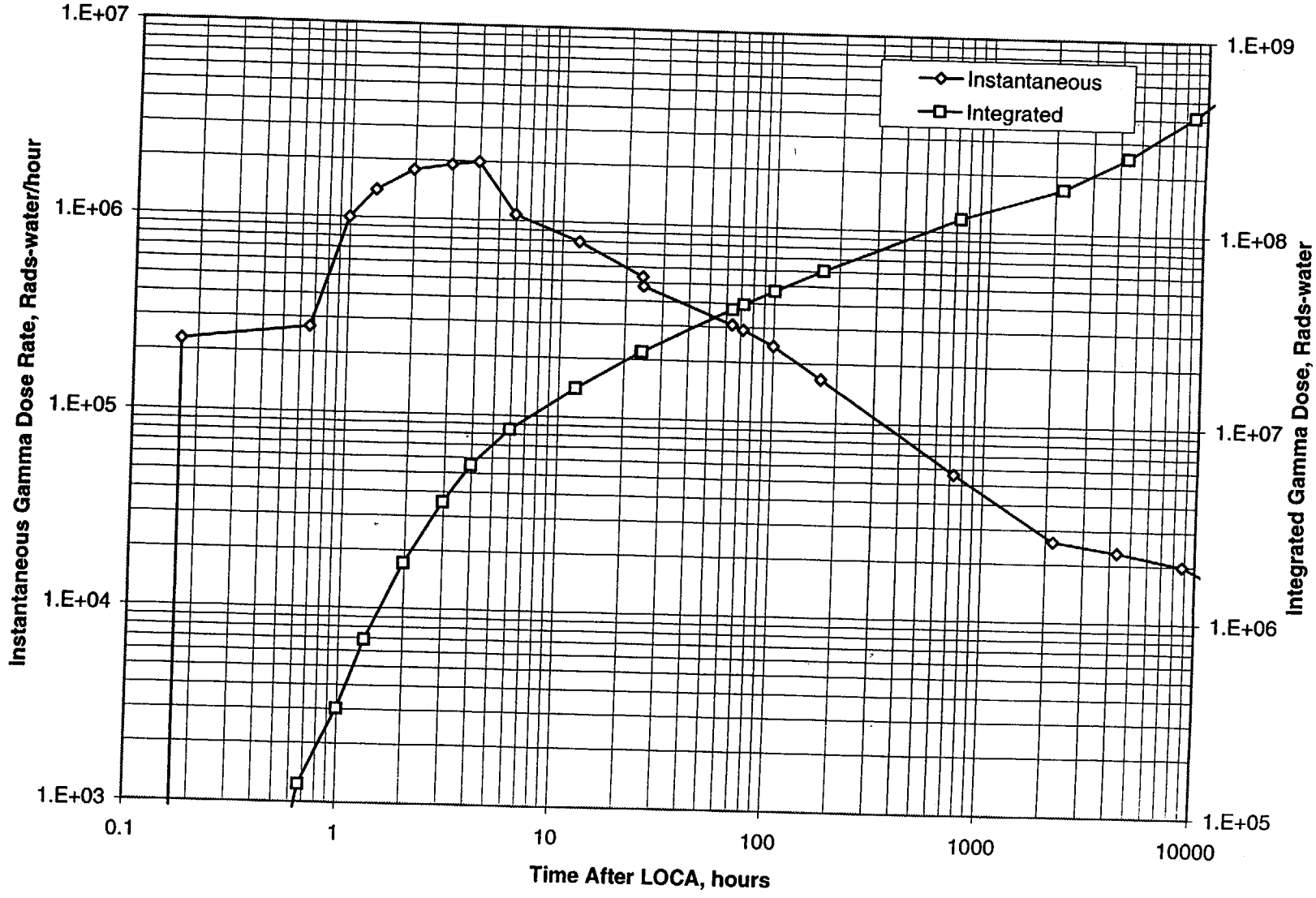


Figure 8
Beta Dose in Containment Sump Water After Severe Accident

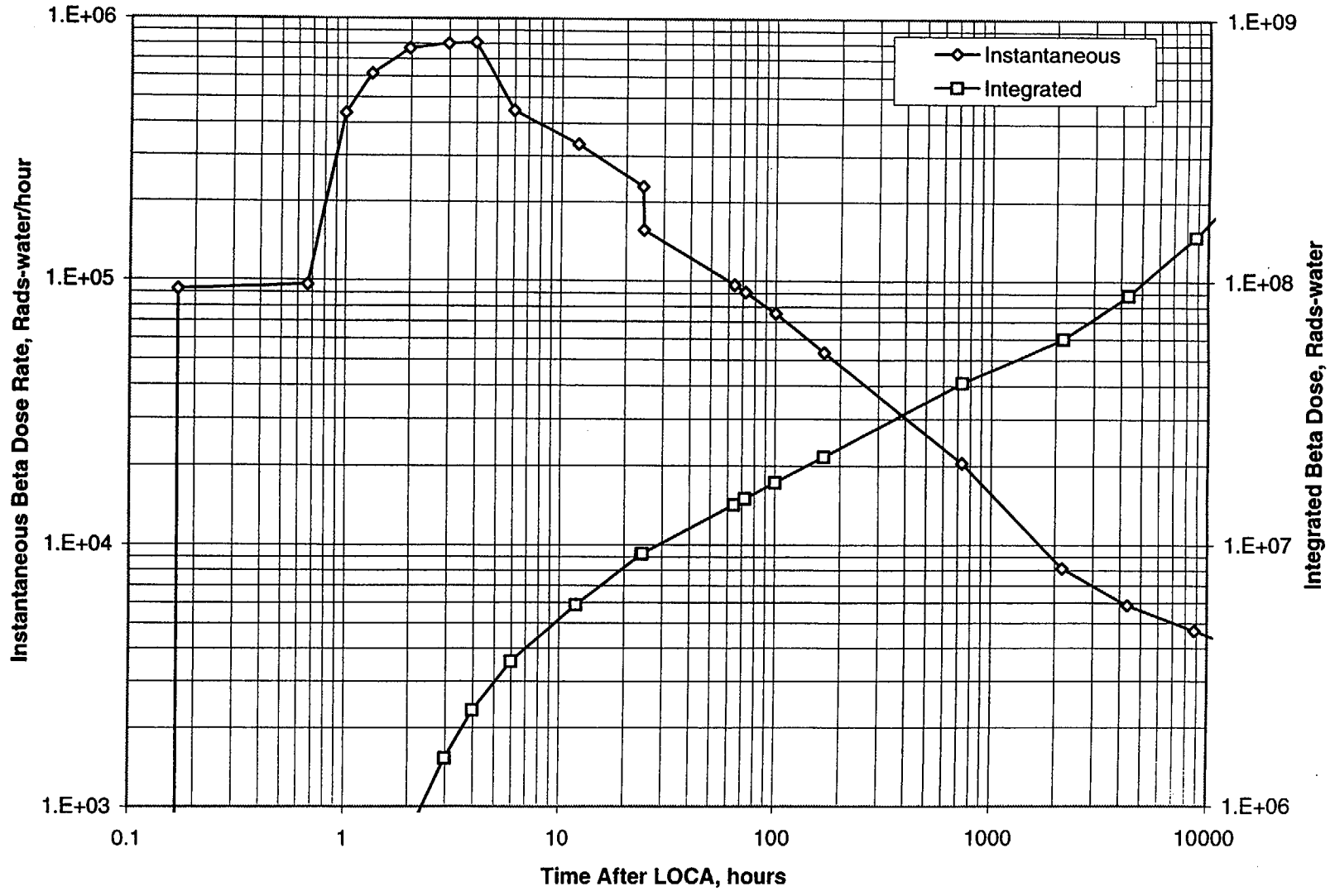


Figure 9
Integrated Gamma Dose in Containment Air

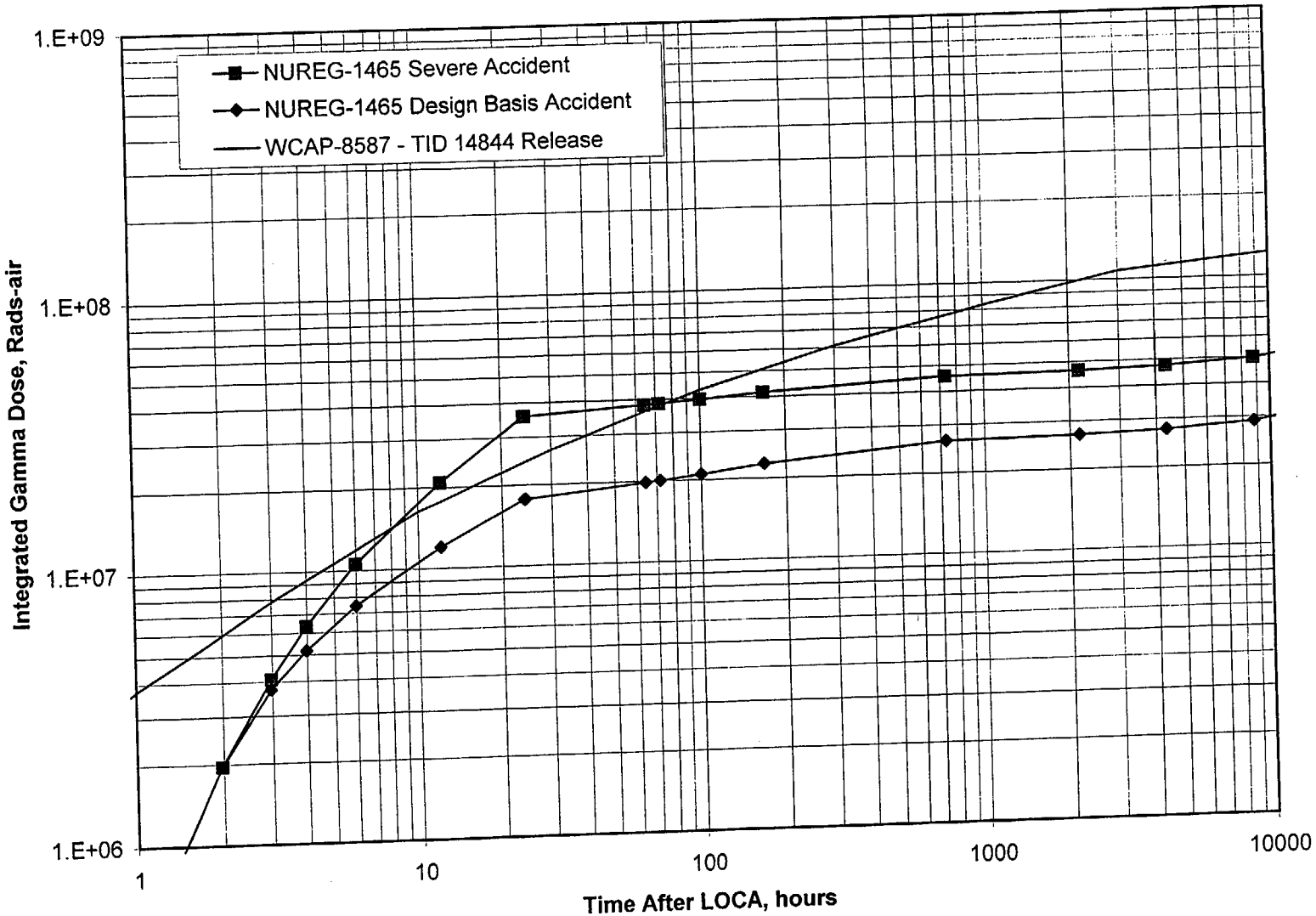


Figure 10
Integrated Beta Dose in Containment Air

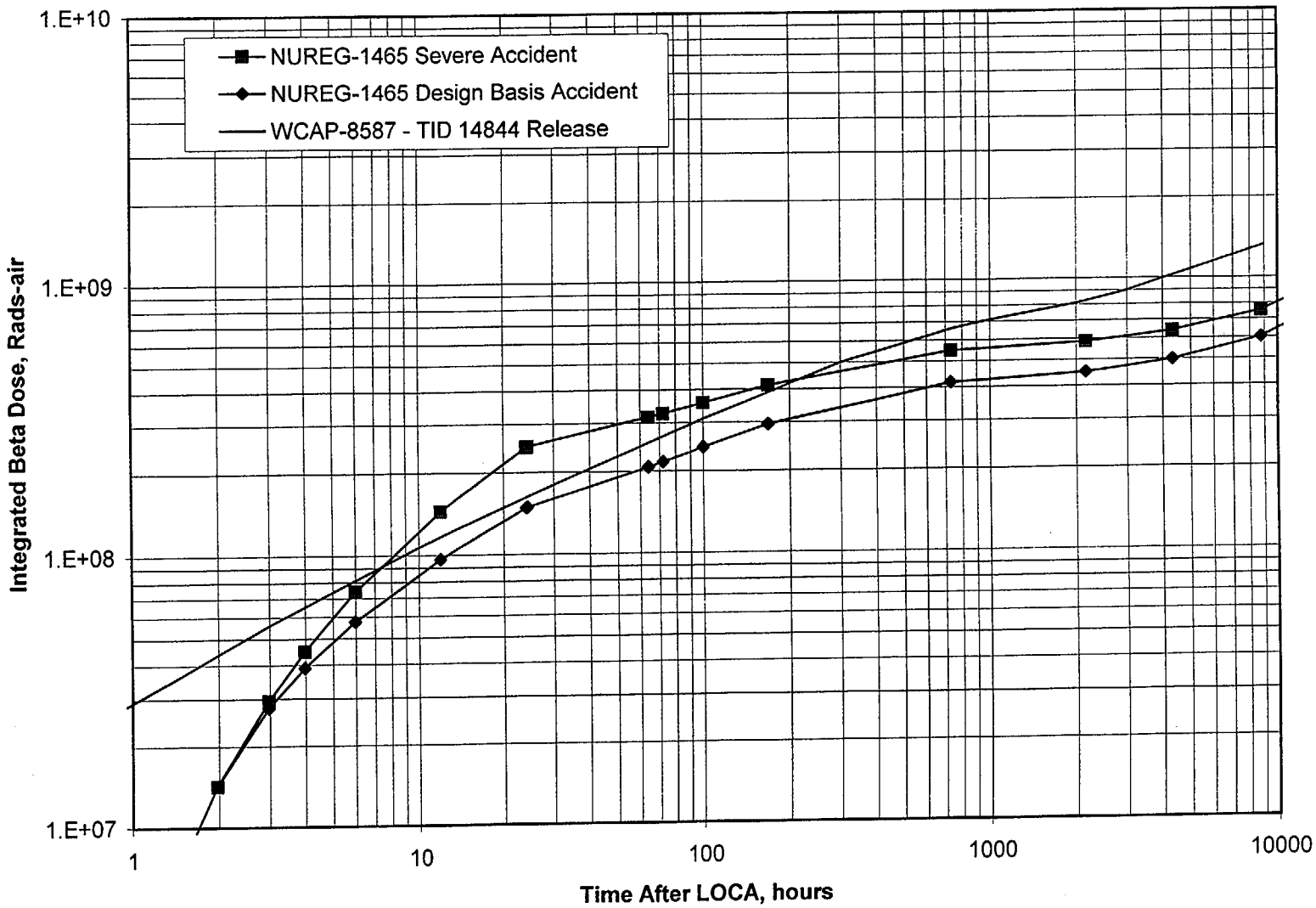
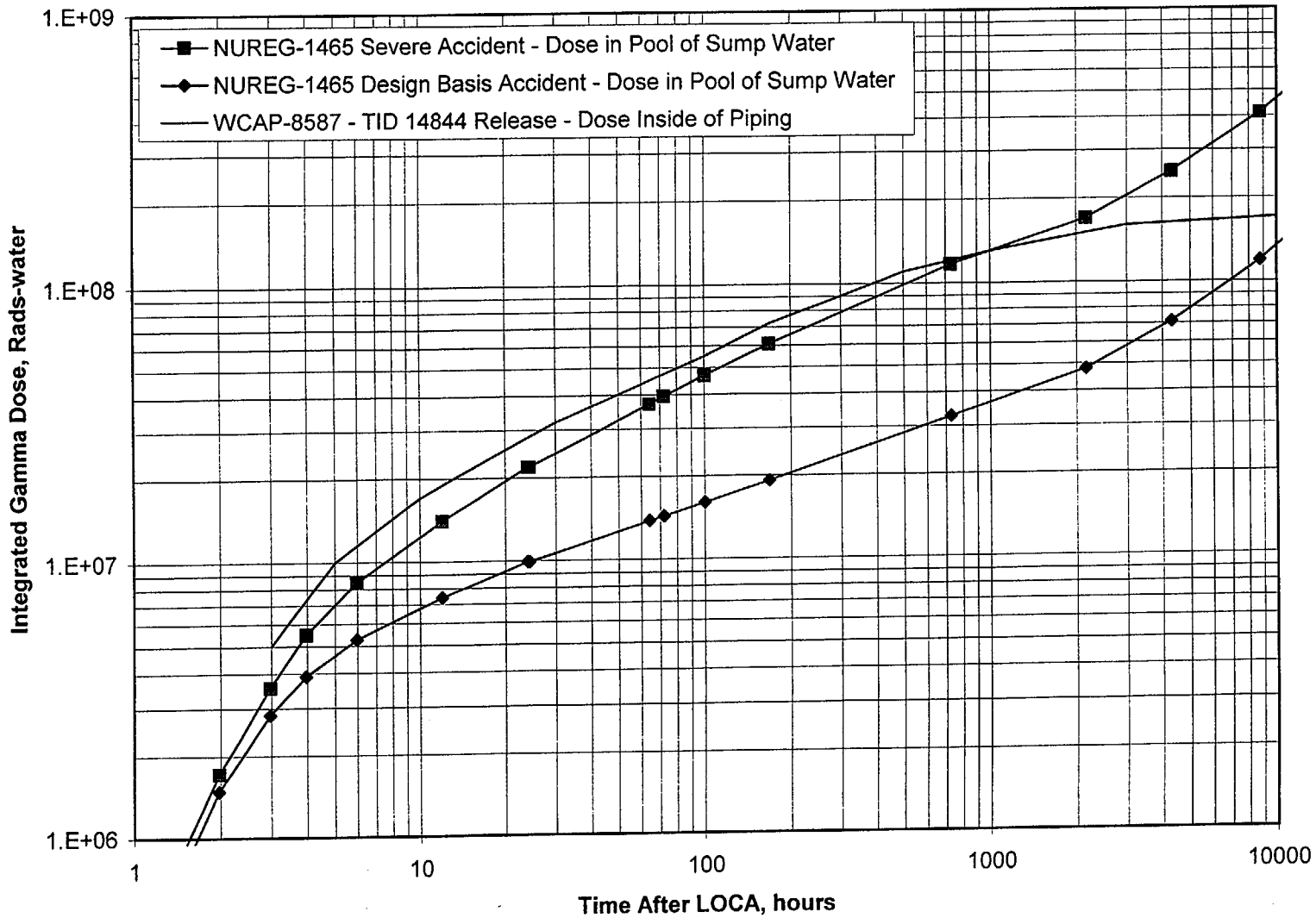


Figure 11
Integrated Gamma Dose in Containment Sump Water



ATTACHMENT 9 TO C0600-13

COMMITMENTS

The following table identifies those actions committed to by Indiana Michigan Power Company (I&M) in this submittal. Other actions discussed in the submittal represent intended or planned actions by I&M. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Date
I&M will develop and implement administrative controls for Unit 1 in accordance with Generic Letter (GL) 91-18 to support plant restart with the TID-14844 accident source term.	Prior to Mode 4 entry for Unit 1
I&M will install an additional normal intake air damper in series with the existing normal intake damper and an additional emergency intake air damper in parallel with the existing emergency damper for Unit 1.	Prior to Mode 4 entry for Unit 1
I&M will submit an analysis for the offsite dose for NRC review and approval. This submittal will allow I&M to fully implement the alternative source term from NUREG-1465 and change the design basis assumed in the Updated Final Safety Analysis Report. This analysis will include verification of the χ/Q value contained in the revised control room dose analysis.	Prior to the next refueling outage on Unit 2
I&M will continue to conduct cycle specific analyses of Unit 2 locked rotor events as needed to demonstrate that the event would not result in control room doses that exceed the 5 rem TEDE limits of 10 CFR 50.67.	Prior to the start of each fuel cycle if needed
I&M will provide a supplement to this amendment request addressing the locked rotor event for Unit 1.	Prior to Mode 2 entry for Unit 1
I&M will determine the effect of postulated single failures of the emergency inlet dampers on the doses for the analyzed accidents and will provide a supplement to this submittal if the doses differ significantly from the values given in Attachment 6.	Prior to Mode 2 entry for Unit 1
The testing methodology recommended in GL 99-02 will be incorporated in the next laboratory testing of the activated charcoal for the control room emergency ventilation system for Unit 1. The testing will be done in accordance with ASTM D3803-1989 or the charcoal will be replaced with new charcoal tested in accordance with the new standard.	Prior to Mode 4 entry for Unit 1
The testing methodology recommended in GL 99-02 will be incorporated in the next laboratory testing of the activated charcoal for the engineering safety features ventilation system (ESFVS) for Unit 1. The testing will be done in accordance with ASTM D3803-1989 or the charcoal will be replaced with new charcoal tested in accordance with the new standard.	Prior to Mode 4 entry for Unit 1