FPL Energy Point Beach, LLC, 6610 Nuclear Road, Two Rivers, WI 54241

FPL Energy
Point Beach Nuclear Plant

October 1, 2007

NRC 2007-0040
10 CFR 50.90

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Point Beach Nuclear Plant Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

License Amendment Request 241
Alternative Source Term

Pursuant to 10 CFR 50.90, FPL Energy Point Beach, LLC (FPLE-PB) requests to amend Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant (PBNP), Units 1 and 2, respectively. FPLE-PB proposes to revise the PBNP licensing bases to adopt the alternative source term (AST) as described in 10 CFR 50.67 following the guidance provided in Regulatory Guide (RG) 1.183.

Enclosure 1 provides a description of the proposed changes, supporting justification, the No Significant Hazards Consideration and Environmental Consideration. Enclosure 2 provides proposed Technical Specification (TS) changes needed to implement AST. Associated TS Bases pages are also provided in Enclosure 2 for Staff information. Enclosure 3 provides the technical evaluation supporting this license amendment request. Enclosure 4 provides the Regulatory Issues Summary (RIS) 2006-04 resolution matrix. Enclosure 5 provides the Regulatory Guide (RG) 1.183 compliance matrix. Enclosure 6 provides ARCON96 meteorological data files from September 2000 to October 2005 and drawings on a CD in support of this license amendment request. This CD also includes applicable FSAR Chapter 14 accident analyses.

Approval of this proposed license amendment is required to support planned actions to address Generic Letter 2003-01, Control Room Habitability.

Summary of Commitments

This letter completes the following commitment made in the NMC response to Generic Letter 2003-01 dated September 5, 2007 (ADAMS Accession No. ML072490030):

> NMC will submit a license amendment request to the NRC revising the current accident analysis for PBNP to demonstrate compliance with the dose limits of 10 CFR 50, Appendix A, GDC-19, using the Alternative Source Term by October 1, 2007. As part of this submittal, the post accident reliance on KI (potassium iodide) for control room staff will be addressed.

CD Noncompliant with NRC Requirements... Paper Version processed per J. Cushing ... CD forward to NRC File Center.

A001
A102
NRR

This letter contains four new commitments:

1. PBNP will implement the following plant modifications to support the assumptions used in the radiological dose analysis:

    - A new control room emergency filtration system Mode 5 will be established, including operating procedure changes, with combined filtered outside air and filtered recirculation air.

    - Control room shielding modifications will be made outside the control room envelope.

    - Modifications to the residual heat removal and containment spray, and their support systems, will be implemented to support operation of containment spray during containment sump recirculation.

2. The PBNP emergency operating procedures (EOPs) will be revised to direct continued containment spray during emergency core cooling system recirculation if containment radiological conditions and/or core damage indicates containment spray is required.

3. A supplement to this license amendment request will be submitted by January 15, 2008, to enable NRC staff review of the modifications when the final designs for the modifications to the residual heat removal and containment spray, and their support systems are complete.

4. The Technical Specification Bases and FSAR descriptions will be revised as appropriate to support the proposed license amendment request for the implementation of the AST and revision to the associated TS.

FPLE-PB requests approval of the proposed license amendment by October 1, 2008, with the amendments being implemented following the Unit 2 refueling outage in the fall of 2009.

FPLE-PB has evaluated the proposed amendment and has determined that it does not involve a significant hazards consideration pursuant to 10 CFR 50.92. The PBNP Plant Operations Review Committee has reviewed the proposed license amendment request.

In accordance with 10 CFR 50.91, a copy of this application, with Enclosures 1, 2, 3, 4, and 5 is being provided to the designated Wisconsin Official.

Please address questions regarding this license amendment request to Mr. Larry Peterson at 920/755-7441.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: October 1, 2007

Very truly yours,
FPL Energy Point Beach, LLC

FOR

Dennis L. Koehl
Site Vice President

Enclosures (6)

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW (less Enclosure 6)

ENCLOSURE 1

LICENSE AMENDMENT REQUEST 241
REGULATORY ASSESSMENT OF THE PROPOSED IMPLEMENTATION OF THE ALTERNATIVE RADIOLOGICAL SOURCE TERM METHODOLOGY

POINT BEACH NUCLEAR PLANT

16 Pages Follow

ENCLOSURE 1

LICENSE AMENDMENT REQUEST 241
REGULATORY ASSESSMENT OF THE PROPOSED IMPLEMENTATION OF THE ALTERNATIVE RADIOLOGICAL SOURCE TERM METHODOLOGY

POINT BEACH NUCLEAR PLANT

1. SUMMARY DESCRIPTION

Introduction

This submittal requests to amend Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant (PBNP), Units 1 and 2, respectively. Pursuant to the requirements of 10 CFR 50.90 and 10 CFR 50.67, FPL Energy Point Beach, LLC (FPLE Point Beach) proposes to amend Appendix A of Facility Operating Licenses DPR-24 and DPR-27, Technical Specifications. This request incorporates a revision to the licensing basis of PBNP that supports a full-scope application of an Alternative Source Term (AST) methodology. Proposed Technical Specification (TS) changes, which are supported by the AST design basis accident (DBA) radiological consequence analyses, are included in this application for a license amendment as Enclosure 2.

The current PBNP licensing basis for radiological consequence analysis of accidents discussed in Chapter 14 of the Final Safety Analysis Report (FSAR) is based upon methodologies and assumptions that are primarily derived from Technical Information Document (TID)-14844 and other early guidance.

The AST methodology will modify PBNP's licensing bases by: (1) replacing the current accident source term with an AST as described in 10 CFR 50.67 for DBA radiological consequences, and (2) establishing the 10 CFR 50.67 Total Effective Dose Equivalent (TEDE) dose limits as acceptance criteria for the radiological consequences of DBAs.

Regulatory Guide (RG) 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", provides guidance on application of AST in revising the accident source terms used in design basis radiological consequences analyses, as allowed by 10 CFR 50.67. Because of advances made in understanding the timing, magnitude and chemical form of fission product releases from severe nuclear power plant accidents, 10 CFR 50.67 was issued to allow holders of operating licenses to voluntarily revise the traditional accident source term used in the DBA radiological consequence analyses with ASTs.

Evaluation Overview and Objective

As documented in NEI 99-03 and Generic Letter 2003-01, several other nuclear plants performed testing on control room unfiltered air inleakage that demonstrated leakage rates in excess of amounts assumed in the current accident analyses. While the PBNP tracer gas test demonstrated inleakage greater than that assumed in the FSAR analyses, the AST methodology in RG 1.183 and as supplemented by Regulatory Issues Summary 2006-04, is being used to calculate the offsite and control room radiological consequences for PBNP to support the control room habitability program by establishing a conforming set of radiological analyses.

The following limiting FSAR Chapter 14 accidents are analyzed:

14.1.8, Loss of Coolant Flow (Locked Rotor - LR)
14.2.1, Fuel Handling Accident (FHA)
14.2.4, Steam Generator Tube Rupture (SGTR)
14.2.5, Rupture of a Steam Pipe (Main Steam Line Break, MSLB)
14.2.6, Rupture of a Control Rod Mechanism Housing – RCCA Ejection (CRDE)
14.3.5, Large Break Loss of Coolant Accident (LOCA)
14.3.6, Reactor Vessel Head Drop Accident (RVHD)

The fuel handling accident is approved to use the AST methodology. It is included in this amendment to present the change in the dose to the control room due to a proposed control room heating, ventilation and air conditioning (HVAC) emergency mode change, change in the source term, and a change in the atmospheric dispersion factors for the CR.

2. PROPOSED CHANGES

Licensing Basis and Technical Specification Changes

FPLE proposes to revise the PBNP licensing basis to implement the AST, described in RG 1.183, through reanalysis of the radiological consequences of the FSAR Chapter 14 accidents listed in Section 1 above. As part of the full implementation of this AST, the following changes are incorporated in the analysis:

- The AST methodology is adopted for the composition, magnitude, chemical form and timing of radiation releases, as well as accident-specific modeling for all radiological DBAs presented in the PBNP FSAR;
- The TEDE acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11;
- Atmospheric dispersion factors for the control room intake are reanalyzed for existing pathways using ARCON96;
- New values for control room unfiltered air inleakage are assumed. The AST methodology as established in RG 1.183 is being used to calculate the offsite and control room radiological consequences for PBNP to support the control room habitability program;
- The control room ventilation system is operating with filtered recirculation in addition to a filtered fresh air intake;
- Credit is taken for future shielding modifications to the control room;
- A reduced value in the allowable dose equivalent (DE) I-131 concentrations in the primary and secondary systems is used;
- A reduced containment leakage is modeled;

- Credit is taken for the use of containment spray while on emergency core cooling system (ECCS) recirculation LOCA;

- A factor of two increase is applied to the ECCS leakage limit for control room habitability radiological analysis;

- Flashing fractions are applied to the SGTR break flow;

- Credit is not taken for the administration of potassium iodide (KI) to control room personnel;

- Elimination of the requirement that the reactor must be shut down for 100 hours prior to lifting the reactor vessel head based on the revised analyses.

Accordingly, the following changes to the PBNP TS are proposed:

1. TS Section 1.1, Definitions. The definition of L_a used in the containment leakage rate testing program (TS 5.5.15). L_a is reduced from 0.4% of containment air weight per day at P_a to 0.2% of containment air weight per day at P_a.

2. TS Section 3.4.16, RCS Specific Activity. The specific activity of the reactor coolant is revised from DOSE EQUIVALENT I-131 >0.8 μCi/gm to >0.5 μCi/gm in the Conditions and in Surveillance Requirement SR 3.4.16.2.

3. TS Section 3.7.9, Control Room Emergency Filtration System (CREFS). SR 3.7.9.6 is revised for clarification of Mode 5 operation.

4. TS Section 3.7.13, Secondary Specific Activity. The specific activity of the secondary coolant is revised from $\leq$1.00 to $\leq$0.1 μCi/gm DOSE EQUIVALENT I-131 in the LCO and Surveillance Requirements.

5. TS Section 5.5.15, Containment Leakage Rate Testing Program. The maximum allowable containment leakage rate, L_a at P_a shall be 0.2% of containment air weight per day.

Credited Plant Modifications

Plant operational and equipment requirements to meet PBNP's revised radiological analyses based on adoption of AST have been reviewed. Revisions to the PBNP operational configuration and plant modifications to meet these requirements have been developed through the conceptual design phase. The changes are described below:

1. A change in post-LOCA operational requirements to provide post-LOCA operation of containment spray for four hours in the ECCS recirculation phase.

2. Modification to containment spray (CS) and residual heat removal (RHR) systems to provide throttling capability of RHR and CS, along with backup nitrogen for system valves.

3. A plant modification to add radiological shielding in areas of the PBNP control room to ensure control room habitability requirements are maintained.

4. A plant modification to CREFS for an additional operational mode that provides combined outside and return air.

To support specific assumptions used in the AST radiological consequence analyses, plant modifications are necessary. A description of the conceptual designs of these modifications follows:

1. A new CREFS Mode 5 will be established. This change will provide for a combination of filtered outside air and filtered recirculation. All ductwork modifications are expected to be outside of the control room envelope. The new mode will be actuated by either a high radiation signal from control room noble gas or area monitors, a containment isolation signal or manually. The new system will provide a maximum of 2500 cfm of outside air and a minimum of 1955 cfm of return air. An additional balancing damper on the outside air duct is planned. The ability to operate in the current modes 3 and 4 will be retained. Tracer gas testing in the proposed mode 5, which is a combination of current modes 3 and 4, has been previously tested with satisfactory results.

2. The increased shine doses to the control room operators will be reduced by shielding modifications made to the control room outside of the control room envelope. These modifications will add a shield to cover the east window of the control room, a concrete vestibule at the south entrance to the control room, a concrete layer will be added to the portion of the HVAC floor above the control room and two windows will be blocked off in the Operations office outside of the control room.

3. The AST LOCA dose analysis assumes CS is operated for four hours while in the ECCS recirculation phase. In order to operate the CS system during the ECCS recirculation phase, the ability to throttle both CS and the RHR flow is required to maintain adequate RHR pump net positive suction head (NPSH). Modifications to the RHR system include the addition of a nitrogen backup system and modifications of the control cables for the RHR flow throttle valves.

4. Modifications are required to the CS discharge valves and wiring for the valve control switches at the main control board to allow the valves to be placed in intermediate positions.

5. Associated with the modifications to the RHR and CS systems, an emergency operating procedure (EOP) change will be needed to operate CS during ECCS recirculation. The modifications to provide for RHR and CS throttling will provide the capability to operate a single train of RHR for core cooling and CS pump suction while maintaining adequate NPSH for the RHR pumps.

Current FSAR radiological accident analyses do not take credit for operation of the CS system in the ECCS recirculation phase.

The dose projections prepared in support of this submittal assume that containment spray is maintained throughout the injection phase of a LOCA and continued during the early portions of the recirculation phase with no more than a 20-minute interruption. The ability to maintain spray during the early recirculation phase is essential, as this is the period of highest iodine evolution from a damaged core.

To support this assumption, it will be necessary for PBNP to alter the existing EOPs to direct continued CS while on sump recirculation if containment radiological conditions and/or core damage indicates it is required.

To enhance NRC staff reviews, a supplement to this license amendment request will be submitted by January 15, 2008, when the final designs for the modifications are complete.

3. BACKGROUND

In the past, power reactor licensees typically used U.S. Atomic Energy Commission Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962, as the basis for DBA analysis source terms. The power reactor siting regulation, 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," which contains offsite dose limits in terms of whole body and thyroid dose, references TID-14844.

On December 23, 1999, the NRC published 10 CFR 50.67, "Accident source term," in the Federal Register. This regulation provides a mechanism for licensed power reactors to replace the current accident source term used in DBA analyses with an AST. The direction provided in 10 CFR 50.67 is that licensees who seek to revise their current AST in design basis radiological consequence analyses apply for a license amendment pursuant to 10 CFR 50.90.

RG 1.183 was used in preparing AST DBA radiological consequence analyses for PBNP. The NRC staff prepared RG 1.183 to provide guidance to licensees of operating power reactors on acceptable applications of ASTs; the scope, nature and documentation of associated analyses and evaluations; considerations of impacts on analyzed risk; and contents of submittals.

The criteria of 10 CFR 50.67 were used to evaluate the current licensing basis (CLB) DBAs for radiological consequences. The analysis assumptions are consistent with RG 1.183, as presented in Enclosure 5.

4. TECHNICAL EVALUATION SUMMARY

The technical evaluation supporting this submittal is provided in Enclosure 3. A summary of the significant points of the evaluation are provided below.

Radiological Evaluation Methodology

Acceptance Criteria

Offsite and control room doses must meet the guidelines of RG 1.183 and requirements of 10 CFR 50.67. The acceptance criteria for specific postulated accidents are provided in Table 6 of RG 1.183. For analyzed events not addressed in RG 1.183, the basis used to establish the acceptance criteria for the radiological consequences is provided in the discussion of the event in Enclosure 3.

Analysis Input Assumptions

Common analysis input assumptions include those for the control room ventilation system and dose calculation model, direct shine dose, radiation source terms and atmospheric dispersion factors. Event-specific assumptions are discussed in the event analyses contained in Enclosure 3.

PBNP Design Basis Loss-of-Coolant Accident

The control room shielding design is based on the LOCA, which represents the worst case DBA relative to radioactivity releases. The direct shine dose due to contained sources and the external cloud is included in the control room doses reported for the LOCA. The contribution of shine due to other accidents is not explicitly determined for non-LOCA accidents. For the non-LOCA accidents, it is assumed that the shine dose contribution is negligible.

Radiological consequences due to a LOCA are due to a postulated abrupt failure of the main reactor coolant pipe in which the emergency core cooling features fail to prevent the core from experiencing significant degradation such as melting. Activity from the core is released to the containment and from there released to the environment by containment leakage and leakage from the ECCS.

The reanalysis of the LOCA offsite and control room doses for PBNP uses the following RG 1.183 source term characteristics in place of those identified in TID-14844:

- Iodine chemical species
- Fission product release timing
- Fission product release fractions

PBNP Licensing Basis

PBNP was licensed prior to the 1971 publication of 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC). As such, PBNP is not licensed to the Appendix A GDCs. PBNP FSAR Section 1.3 lists the plant-specific GDC to which the plant was licensed. The PBNP GDCs are similar in content to the draft GDC proposed for public comment in 1967.

The following discussion addresses the proposed change with respect to meeting the requirements of the applicable draft design criteria:

- PBNP GDC 17, Monitoring Radioactivity Releases, states that means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients and from accident conditions.

  The proposed changes will have no impact upon how the requirements of PBNP GDC 17 are met.

- PBNP GDC 18, Monitoring Fuel and Waste Storage Areas, states that monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels.

  The proposed changes have no impact upon how the requirements of PBNP GDC 18 are met.

- PBNP GDC 70, Control of Releases of Radioactivity to the Environment, states that the facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity and control shall be justified (a) on the basis of 10 CFR 20 requirements, for both normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence.

  The proposed changes will have no impact on how the requirements of PBNP GDC 70 are currently met. Under accident conditions, radioactive gaseous effluents that may be released into enclosed areas are collected by the ventilation systems and discharged to the plant vent. Permanently installed area radiation monitors and plant vent radioactivity detectors are used to monitor the discharge levels to the environment. In addition, portable radiation monitors are available on site for supplemental surveys. The releases from these accidents have been calculated to be less than 10 CFR 20 and 10 CFR 100 requirements.

Fission Product Inventory and Source Terms

The source term data to be used in performing AST analyses for PBNP are:

- Core Inventory - A new core source term has been calculated for use in the radiological accident analyses. The inventory of the fission products in the reactor core is based on maximum full-power operation of the core at a power level equal to 1683 MWt, and current licensed values of fuel enrichment and burnup. The core mass calculated is 48.0 MTU with an equilibrium cycle length of 17,175 MWD/MTU. The fuel was modeled with an active fuel length of 132 inches with axial blanket regions of six inches in length. The current licensed core power level is 1540 MWt. Although the analyses were performed at the higher power level, this amendment request is not requesting approval for operation at the higher power level. Enclosure 3 provides additional information on the new core source term.

- Reactor Coolant Source Term - For the reactor coolant system, maximum coolant activities obtained during a cycle of operation are calculated. Small cladding defects in fuel are assumed present at initial core loading and uniformly distributed throughout the core. The radiation source is based on 1% of the rods containing fuel defects operating at a core power of 1683 MWt. The RCS activity is determined using a calculated minimum liquid volume to obtain conservative concentrations. Other parameters used in determining coolant inventory include the pertinent information concerning the expected coolant cleanup flow rate, demineralizer effectiveness and volume control tank noble gas stripping behavior. Additional information is contained in Enclosure 3.

- Gap Inventory for Non-LOCA Accidents - The gap fractions listed in Section 3.2 of RG 1.183 serve as a basis for determining available activity in the Control Rod Drive Ejection (CRDE), Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Reactor Vessel Head Drop (RVHD), Fuel Handling Accident (FHA) and Locked Rotor (LR) radiological analyses. RG 1.183 Table 3 Footnote 11 states that the gap fractions are acceptable for light water reactor fuel with a peak rod burnup less than 62,000 MWD/MTU, provided the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54,000 MWD/MTU.

For the fuel handling accident (FHA), higher gap fractions were applied for the fraction of the fuel that does not meet the limits on burnup and linear heat generation rate, following the method approved by the NRC for Kewaunee (ML070430020). The gap fractions are those from Safety Guide 25 with the value for I-131 adjusted consistent with the recommendation of NUREG/CR-5009. The detail gap inventory for non-LOCA accident is addressed in Section 3.3 of Enclosure 3.

Atmospheric Relative Concentrations

The atmospheric dispersion (χ/Q) values for the PBNP exclusion area boundary (EAB) and the low population zone (LPZ) are those from the CLB. These values were developed from the guidance provided in RG 1.145 and meteorological data collected at the site's primary tower from January 1, 1991, through December 31, 1993. The offsite χ/Q values are presented in Table 3 of Enclosure 3 and represent the maximum sector χ/Q values.

Current Licensed Power Level and AST Modeling

The current licensed maximum power level is 1540 MWt. The analyses in this report model a maximum power of 1683 MWt. Although the analyses were performed at a higher power level, this license amendment is not requesting approval for use of the higher power.

Methodologies

The AST analyses performed for PBNP use assumptions and models defined in RG 1.183 to provide appropriate and prudent safety margins. Except as otherwise stated, credit is taken for engineered safety features (ESF) and other appropriately qualified, safety-related accident mitigation features.

Analysis Conservatisms

In order to maximize the resulting doses and to provide margin allowance, conservatisms have been used while preparing this submittal. A discussion of these conservatisms is provided for each event in the corresponding section in Enclosure 3.

Conclusion

Full implementation of the alternative source term methodology, as defined in RG 1.183 and Regulatory Issue Summary 2006-04, into the DBA analysis will support control room habitability. Analyses of the dose consequences of the loss-of-coolant accident, locked rotor, fuel handling accident, steam generator tube rupture, main steam line break, rod cluster control assembly ejection, and reactor vessel head drop accident have been performed using the RG 1.183 methodology. The analyses used assumptions consistent with the proposed changes in the PBNP licensing basis and the calculated doses do not exceed the defined acceptance criteria.

Summary of Results

Results of the PBNP radiological consequence analyses using the AST methodology and the corresponding allowable control room unfiltered air inleakage are summarized in Enclosure 3. Enclosure 3 explains these results and acceptance criteria in more detail. Enclosure 3 supports a maximum allowable control room unfiltered air inleakage of 105 cfm.

5. REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

In accordance with the requirements of 10 CFR 50.90, FPL Energy Point Beach, LLC (FPLE-PB) the licensee, hereby requests amendments to Facility Operating Licenses DPR-24 and DPR-27, for Point Beach Nuclear Plant, Unit 1 and Unit 2.

The standards used to arrive at a determination that a request for an amendment involves a no significant hazards consideration are included in 10 CFR 50.92, which states that no significant hazards considerations are involved if the operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or

consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

FPLE-PB has evaluated the proposed amendment in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of the Point Beach Nuclear Plant in accordance with the proposed amendment presents no significant hazards. The FPLE-PB evaluation against each of the criteria in 10 CFR 50.92 follows.

1. *Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?*

   Response: No.

   The results of the applicable radiological design basis accident (DBA) re-evaluation demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory limits and guidance provided by the NRC in 10 CFR 50.67 and Regulatory Guide 1.183 for alternative source term (AST) methodology. The AST is an input to calculations used to evaluate the consequences of an accident and does not by itself affect the plant response or the actual pathway of the activity released from the fuel. It does, however, better represent the physical characteristics of the release such that appropriate mitigation techniques may be applied.

   The change from the original source term to the new proposed AST is a change in the analysis method and assumptions and has no effect on accident initiators or causal factors that contribute to the probability of occurrence of previously analyzed accidents. Use of an AST to analyze the dose effect of DBAs shows that regulatory acceptance criteria for the new methodology continues to be met. Changing the analysis methodology does not change the sequence or progression of the accident scenario.

   The proposed Technical Specification changes reflect the plant configuration that will support implementation of the AST analyses. The equipment affected by the proposed changes is mitigative in nature and relied upon after an accident has been initiated. The operation of various filtration systems, the residual heat removal and the containment spray system, including associated support systems, has been considered in the evaluations for these proposed changes. While the operation of these systems does change with the implementation of an AST, the affected systems are not accident initiators, and application of the AST methodology itself, is not an initiator of a design basis accident.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   As described in Item 1 above, the changes proposed in this license amendment request involve the use of a new analysis methodology and related regulatory acceptance criteria. The proposed Technical Specification changes reflect the plant configuration that will

support implementation of the new methodology. No new or different accidents result from utilizing the proposed changes. Although the proposed changes require modifications to the control room emergency ventilation system, as well as modifications to the residual heat removal system and containment spray system, these changes will not initiate a new or different kind of accident since they are related to system capabilities that provide protection from accidents that have already occurred. As a result, no new failure modes are being introduced that could lead to different accidents. These changes do not alter the nature of events postulated in the Updated Final Safety Analysis Report nor do they introduce any unique precursor mechanisms.

Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. *Does the proposed amendment involve a significant reduction in a margin of safety?*

Response: No.

As described in Item 1, the changes proposed in this license amendment involve the use of a new analysis methodology and related regulatory acceptance criteria. The proposed Technical Specification changes reflect the plant configuration that will support implementation of the new methodology. Safety margins and analytical conservatisms have been evaluated and have been found to be acceptable. The analyzed events have been carefully selected and, with plant modifications, margin has been retained to ensure that the analyses adequately bound postulated event scenarios. The proposed changes continue to ensure that the dose consequences of DBAs at the exclusion area and low population zone boundaries and in the control room are within the corresponding acceptance criteria presented in RG 1.183 and 10 CFR 50.67. The margin of safety for the radiological consequences of these accidents is provided by meeting the applicable regulatory limits, which are set at or below the 10 CFR 50.67 limits. An acceptable margin of safety is inherent in these limits.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, FPLE concludes that the proposed change presents no significant hazards under the standards set forth in 10 CFR 50.92, and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

General Design Criteria

PBNP was licensed prior to the 1971 publication of 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC). As such, PBNP is not licensed to the Appendix A GDCs. PBNP Final Safety Analysis Report (FSAR), Section 1.3, lists the plant-specific GDCs to which the plant was licensed. The Point Beach GDCs are similar in content to the draft GDCs proposed for public comment in 1967.

The revised analyses for the DBAs identified in Section 1 are based upon 10 CFR 50.67 and use the regulatory guidance of RG 1.183. The analyses demonstrate compliance with these regulatory guides and criteria. Use of the new analysis method replaces 10 CFR 100 as the applicable dose acceptance criteria for all DBAs.

GDC 19 requires that holders of an operating license using an AST under 10 CFR 50.67 shall meet the requirements of the criterion by ensuring the radiation exposures to control room occupants shall not exceed 5 rem Total Effective Dose Equivalent (TEDE) dose. The analysis provided to support the requested changes demonstrates that this requirement is met.

FPLE has determined that the proposed changes do not require any exemptions or relief from regulatory requirements. The proposed use of an AST to evaluate the consequences of a DBA results in a change to the existing licensing basis analysis described in the FSAR. In accordance with 10 CFR 50.71, FPLE will update the FSAR to reflect the proposed new analysis method. The changes to the Technical Specifications incorporate assumptions used in the new analysis.

Compliance with PBNP GDC 17, 18 and 70 is described in Section 4.

FSAR Accident Analysis Compliance

FPLE proposes to revise the PBNP licensing basis to implement the AST described in RG 1.183, through reanalysis of the radiological consequences of the following limiting FSAR Chapter 14 accidents:

14.1.8, Loss of Coolant Flow (Locked Rotor - LR)
14.2.1, Fuel Handling Accident (FHA)
14.2.4, Steam Generator Tube Rupture (SGTR)
14.2.5, Rupture of a Steam Pipe (Main Steam Line Break, MSLB)
14.2.6, Rupture of a Control Rod Mechanism Housing – RCCA Ejection (CRDE)
14.3.5, Large Break Loss of Coolant Accident (LOCA)
14.3.6, Reactor Vessel Head Drop Accident (RVHD)

As part of the full implementation of this AST, the following changes are assumed in the analysis:

- The AST methodology is adopted for the composition, magnitude, chemical form and timing of radiation releases, as well as accident-specific modeling for all radiological DBAs presented in the PBNP FSAR;

- The TEDE acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11;

- Atmospheric dispersion factors for the control room intake are reanalyzed for existing pathways using ARCON96;

- New values for control room unfiltered air inleakage are assumed. The AST methodology as established in RG 1.183 is being used to calculate the offsite and control room radiological consequences for PBNP to support the control room habitability program;

- The control room ventilation system is operating with filtered recirculation in addition to a filtered fresh air intake;
- Credit is taken for future shielding modifications to the control room;
- A reduced value in the allowable dose equivalent (DE) I-131 concentrations in the primary and secondary systems is used;
- A reduced containment leakage is modeled;
- Credit is taken for the use of containment spray while on emergency core cooling system (ECCS) recirculation Loss-of-Coolant Accident (LOCA);
- A factor of two increase is applied to the ECCS leakage limit for control room habitability radiological analysis;
- Flashing fractions are applied to the SGTR break flow;
- Credit is not taken for the administration of potassium iodide (KI) to control room personnel;
- Elimination of the requirement that the reactor must be shut down for 100 hours prior to lifting the reactor vessel head based on the revised analyses.

RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

Enclosure 5 provides a compliance matrix describing how the guidance of RG 1.183 is met for this submittal. There are no exceptions being proposed.

Environmental Qualification

Section C.6 of the RG 1.183 discusses the position on performance of required environmental qualification (EQ) analyses with respect to AST and Technical Information Document, TID-14844 source term assumptions. Generic Safety Issue (GSI) 187 has since been resolved. The NRC staff concluded that there is no clear basis for a requirement to modify the design basis for equipment qualification to adopt AST since there would be no discernible risk reduction associated with such a requirement. Therefore, this license amendment request does not propose to modify the EQ design basis to adopt AST. The PBNP EQ analysis will continue to be based upon TID-14844 assumptions at this time.

Emergency Plan

An evaluation of the change has been performed in accordance with the requirements of 10 CFR 50.54(q) to assess if the proposed changes represent a potential decrease in the effectiveness of the PBNP Emergency Plan. The evaluation concluded that the effectiveness of the Emergency Plan is not affected by these changes.

NUREG-0737 Post-Accident Access Shielding and Sampling Capabilities

PBNP NUREG-0737 post-accident access basis calculations consider 30-day durations for vital area dose rates. Post-accident mission doses associated with actions defined in the PBNP emergency operating procedures are based upon estimates of required mission times and area dose rates from the shielding review study. In the resolution of GSI 187, the NRC staff indicated that for exposure to containment atmosphere, the TID-14844 source term and the gap and in-vessel releases in the AST produced similar integrated doses and that for exposure to sump water, the integrated doses calculated with the AST only exceeded those calculated with TID-14844 after 42 days for a pressurized water reactor.

Additionally, in response to Item Items II.B.2.2 and III.D.3.4, shielding was installed at the C-59 control panel in the PBNP primary auxiliary building, at two 480 V motor control centers (1B32 and 2B32), and at wall penetrations between the control building and the primary auxiliary building. Portable shielding was also provided for the control room windows. In support of this license amendment, plant modifications will be performed to replace the control room portable shielding with permanent shielding.

PBNP has maintained the post-accident sampling systems and has not adopted NRC-approved TSTF-366 as permitted by 65 FR 65018 dated October 31, 2000. Therefore, there is no risk reduction associated with calculation of post-accident access doses via adoption of AST.

Conclusion

Therefore, based upon the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6. ENVIRONMENTAL CONSIDERATION

Adoption of the AST and associated Technical Specification changes, which implement certain conservative assumptions in the AST analyses, will not result in physical changes to the plant that could significantly alter the type or amount of effluents that may be released off-site. No changes to operational parameters that could affect effluent releases have been proposed.

The implementation of the AST has been evaluated in revisions to the analysis of the limiting DBAs at PBNP. Based upon the results of these analyses, it has been demonstrated that with the proposed change, the dose consequences of the limiting event are within the regulatory requirements specified by the NRC for use with the AST (e.g., 10 CFR 50.67 and 10 CFR 50, Appendix A, GDC 19, 10 CFR 20). Thus, there will be no significant increase in either individual or cumulative occupational radiation exposure.

FPLE Point Beach has determined that operation with the proposed amendment would not result in any significant change in the types, or significant increases in the amounts, of any effluents that may be released offsite, nor would it result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible

for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed amendment.

7. PRECEDENTS

NRC has previously approved implementation of the AST methodology at a number of other nuclear power stations including, but not limited to the following:

- Surry Power Station Units 1 and 2, dated March 8, 2002 (ML020710159)
- River Bend Station Unit 1, dated March 14, 2003 (ML030760746)
- Oconee Units 1, 2 and 3, dated June 1, 2004 (ML041540097)
- H.B. Robinson Steam Electric Plant Unit 2, dated September 24, 2004 (ML042680089)
- Seabrook Station Unit 1, dated February 24, 2005 (ML050320373)
- Waterford Unit 3, dated March 29, 2005 (ML050890248)
- Vermont Yankee, dated March 29, 2005 (ML041280490)
- North Anna Power Station Units 1 and 2, dated June 15, 2005 (ML051590510)
- Catawba Station Units 1 and 2, dated September 30, 2005 (ML052730312)
- Salem Units 1 and 2, dated February 17, 2006 (ML060040322)
- Byron Station Units 1 and 2, and Braidwood Units 1 and 2, dated September 8, 2006 (ML062340420)
- Dresden Units 2 and 3, and Quad Cities Units 1 and 2, dated September 11, 2006 (ML062070290)
- Millstone Power Station Unit No. 3, dated September 15, 2006 (ML061990135)
- Columbia Generating Station, dated November 27, 2006 (ML062610440)
- Monticello Nuclear Generating Station, dated December 7, 2006 (ML062790015)
- McGuire Nuclear Station Units 1 and 2, dated December 22, 2006 (ML063100406)
- San Onofre Nuclear Generating Station Units 2 and 3, dated December 29, 2006 (ML063400359)

8. REFERENCES

8.1 NRC Generic Letter (GL) 2003-01, Control Room Habitability, dated June 12, 2003 (ML031620248)

8.2 NMC 60-Day Response to GL 2003-01, dated August 11, 2003 (ML032580984)

8.3 NMC Supplemental Response to GL 2003-01, dated December 5, 2003 (ML042820156)

8.4 NMC Supplemental Response to GL 2003-01, dated September 29, 2004 (ML042820156)

8.5 NMC Supplemental Response to GL 2003-01, dated August 22, 2005 (ML052420480)

8.6 NMC Supplemental Response to GL 2003-01, dated December 8, 2006 (ML063420598)

8.7 NRC letter to NMC dated January 5, 2007 (ML063480040)

8.8 NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ML00370137)

8.9 J.J. DiNunno et al., Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission, March 23, 1962

8.10 L. Soffer et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, USNRC, February 1995

8.11 65 FR 065018, dated October 31, 2000, Elimination of Requirements for a Post-Accident Sampling System (PASS) and TSTF-366 with same title, dated July 21, 2000 (ML003734003)

8.12 Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 7, 2006 (ML053460347)

8.13 NRC letter to Kewaunee Power Station, dated March 8, 2007 (ML070430020)

8.14 NMC letter, Point Beach, Units 1 and 2 - Request for Review of Heavy Load Analysis, dated July 24, 2005 (ML052140556)

8.15 NRC letter to Point Beach Nuclear Plant, Point Beach Units 1 and 2 - Issuance of Amendment RE: Incorporation of Reactor Vessel Head Drop Accident Analysis Into the Final Safety Analysis Report, dated September 23, 2005 (TAC Nos. MC7650 and MC7651) (ML052560089)

ENCLOSURE 2

LICENSE AMENDMENT REQUEST 241

MARKED UP TECHNICAL SPECIFICATION AND TECHNICAL SPECIFICATION BASES CHANGES

POINT BEACH NUCLEAR PLANT

42 Pages Follow

1.1 Definitions

L_a	The maximum allowable primary containment leakage rate, L_a, shall be 0.~~4~~2% of primary containment air weight per day at the peak design containment pressure (P_a).
LEAKAGE	LEAKAGE shall be: a. Identified LEAKAGE 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE); b. Unidentified LEAKAGE All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; c. Pressure Boundary LEAKAGE LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total channel step

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 >0.~~8~~5 μCi/gm.	----------Note---------- LCO 3.0.4.c is applicable. ---------- A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1. AND A.2 Restore DOSE EQUIVALENT I-131 to within limit.	 Once per 4 hours 48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/$\bar{E}$ μCi/gm.	7 days
SR 3.4.16.2	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.~~8~~5 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Operate the CREFS for ≥ 15 minutes.	31 days
SR 3.7.9.2	Perform required CREFS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.9.3	Verify each CREFS emergency make-up fan actuates on an actual or simulated actuation signal.	18 months
SR 3.7.9.4	Verify each CREFS automatic damper in the emergency mode flow path actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.7.9.5	Verify CREFS manual start capability and alignment.	18 months
SR 3.7.9.6	Verify each CREFS emergency make-up fan can maintain a positive pressure ~~of ≥ 0.125 inches water gauge~~ in the control room envelope, relative to the adjacent turbine building during the emergency mode of operation at a ~~makeup~~ flow rate of 4950 cfm ± 10%.	18 months

3.7 PLANT SYSTEMS

3.7.13 Secondary Specific Activity

LCO 3.7.13 The specific activity of the secondary coolant shall be ≤~~1.0~~0.1 μCi/gm DOSE EQUIVALENT I-131.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify the specific activity of the secondary coolant is ≤ ~~1.0~~0.1 μCi/gm DOSE EQUIVALENT I-131.	31 days

5.5 Progams and Manuals

5.5.14 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.15 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995.

b. The peak design containment internal accident pressure, P_a, is 60 psig.

c. The maximum allowable containment leakage rate, L_a at P_a, shall be ~~0.4~~ 0.2% of containment air weight per day.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to the Point Beach design criteria (Ref. 1), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with the Point Beach design criteria (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR ~~100~~ 50.67~~, "Reactor Site Criteria"~~ (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves

BASES

SAFETY LIMIT VIOLATIONS (continued)

and create a potential for radioactive releases in excess of 10 CFR ~~100, "Reactor Site Criteria," limits~~ 50.67. (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, 5, or 6 RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, 5 or 6 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. FSAR, Section 4.1.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR ~~100~~50.67.
5. FSAR, Section 7.2.
6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.
7. FSAR, Section 4.2.

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. The RCS is borated to cold shutdown boron concentration prior to the safety injection system block setpoint being reached during Mode 3.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition; and

c. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

The uncontrolled rod withdrawal transient is terminated by a high power level trip or an OT ΔT trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

BASES

LCO (continued)

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM limit. For MSLB accidents, if the limit is violated, there is a potential to exceed the DNBR limit and to exceed ~~10 CFR 100, "Reactor Site Criteria,"~~ 10 CFR 50.67 as described in Table 6 of Regulatory Guide 1.183 limits (Ref. 4). For the boron dilution accident, if the limit is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 2 with $k_{eff} < 1.0$ and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2 with $k_{eff} \geq 1.0$, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6.

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of 32 gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of 32 gpm and 3.75% boric acid represent typical values and are provided for the purpose of offering a specific example.

BASES

REFERENCES

1. FSAR, Section 3.1.
2. FSAR, Section 14.2.5.
3. FSAR, Section 14.1.4.
4. ~~10 CFR 100~~ Regulatory Guide 1.183, July 2000.
5. FSAR, Sections 14.1.1 and 14.2.6.
6. Westinghouse NSAL 02-014, Steam Line Break During Mode 3.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as specifying LCO's on other reactor system parameters and equipment performance.

The LSSS, defined in this specification as the Allowable Value Setpoints, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);
2. Fuel centerline melt shall not occur; and
3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR ~~100~~ 50.67 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within ~~an acceptable fraction of 10 CFR 100~~ the dose guideline limit of 10 CFR 50.67 as described in Table 6 of Regulatory Guide 1.183 (Ref. 6). Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

to be OPERABLE when the bypass breaker is open or racked out.

These trip Functions must be OPERABLE in MODE 1 or 2 when a Reactor Trip Bypass Breaker is racked in and closed. In MODE 3, 4, or 5, this RPS trip Function must be OPERABLE when a Reactor Trip Bypass Breaker is racked in and closed and the Rod Control System is capable of rod withdrawal.

21. Automatic Trip Logic

The LCO requirement for the RTBs (Functions 18 and 19) and Automatic Trip Logic (Function 21) ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. Each RTB is equipped with a bypass breaker to allow testing of the trip breaker while the unit is at power. The reactor trip signals generated by the RPS Automatic Trip Logic cause the RTBs and associated bypass breakers to open and shut down the reactor.

The LCO requires two trains of RPS Automatic Trip Logic to be OPERABLE. Having two OPERABLE channels ensures that random failure of a single logic channel will not prevent reactor trip. These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RPS trip Functions must be OPERABLE when the RTBs are closed and the Rod Control System is capable of rod withdrawal.

The RPS instrumentation satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.1-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected.

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the unit is outside the safety analysis. Therefore,

REFERENCES

1. FSAR, Chapter 7.
2. FSAR, Chapter 14.
3. IEEE-279-1968.
4. 10 CFR 50.49.
5. DG-I01, Instrument Setpoint Methodology.
6. Regulatory Guide 1.183, July 2000. I

B 3.3 INSTRUMENTATION

B 3.3.5 Control Room Emergency Filtration System (CREFS) Actuation Instrumentation

BASES

BACKGROUND

The CREFS provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. The control room ventilation system normally operates in the normal operating mode (Mode 1). Upon receipt of an actuation signal, the CREFS initiates the emergency ~~make-up~~ (Mode ~~4~~5) mode of operation. The control room ventilation system and its operating modes are described in the Bases for LCO 3.7.9, "Control Room Emergency Filtration System."

The actuation instrumentation consists of containment isolation, noble gas radiation monitor in the air intake and control room area radiation monitor. A containment isolation signal or high radiation signal from either of these detectors will initiate the emergency ~~make-up~~ mode of operation (Mode~~-4~~-5) of the CREFS.

APPLICABLE SAFETY ANALYSES

The CREFS provides airborne radiological protection for control room personnel, as demonstrated by the limiting control room dose analyses for the design basis large break loss of coolant accident. Control room dose analysis assumptions are presented in the FSAR, Section 14.3.5 (Ref. 1).

In MODES 1, 2, 3, and 4, a containment isolation signal or the CREFS radiation monitor actuation signal will provide automatic initiation of CREFS in the emergency ~~make-up~~ mode of operation (Mode~~-4~~ 5) during design basis events which result in significant radiological releases to the environs (e.g. large break loss of coolant accident, steam generator tube rupture, reactor coolant pump locked rotor, etc;).

The CREFS radiation monitor actuation signal also provides automatic initiation of CREFS, in the emergency ~~make-up~~ mode of operation (Mode~~-4~~-5), to assure control room habitability in the event of a fuel handling accident during movement of recently irradiated fuel.

Further Applicable Safety Analysis information for CREFS is contained in the Bases for LCO 3.7.9, "Control Room Emergency Filtration System."

The CREFS actuation instrumentation satisfies Criterion 3 ~~of the NRC Policy Statement~~ of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CREFS is OPERABLE.

1. Control Room Radiation

The LCO requires the control room area (RE-101) and the control room air intake noble gas monitor (RE-235) to be OPERABLE, to ensure that the instrumentation necessary to initiate the CREFS emergency mode ~~make-up~~ (Mode ~~4~~5) is OPERABLE.

Table 3.3.5-1 identifies the Technical Specification Trip Setpoint for the Control Room Area Monitor and Control Room Air Intakes as not applicable (NA). No Analytical Value is assumed in the accident analysis for these functions. The nominal setting required for the Control Room Area Monitor is ~~5~~2 mr/hr and the nominal setting for the Control Room Air Intakes is ~~5~~1E-5 μCi/cc. These nominal settings were developed outside of the setpoint methodology.

2. Containment Isolation

Refer to LCO 3.3.2, Function 3, for all initiating Functions and requirements.

APPLICABILITY

The CREFS Functions must be OPERABLE in MODES 1, 2, 3, 4 and during movement of recently irradiated fuel assemblies.

The Applicability for the CREFS actuation on the ESFAS Safety Injection Functions are specified in LCO 3.3.2. Refer to the Bases for LCO 3.3.2 for discussion of the Safety Injection Function Applicability.

ACTIONS

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.5-1 in the accompanying LCO. The Completion Time(s) of the inoperable Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the containment isolation signal, control room area radiation monitor (RE-101) and the control room intake noble gas monitor (RE-235).

BASES

ACTIONS (continued)

If a Function is inoperable, 7 days is permitted to restore the Function to OPERABLE status from the time the Condition was entered for that Function. The 7 day Completion Time is the same as for inoperable CREFS. The basis for this Completion Time is the same as provided in LCO 3.7.9. If the Function cannot be restored to OPERABLE status, CREFS must be placed in the emergency ~~make-up~~ mode of operation (Mode ~~4~~5). Placing CREFS in the emergency ~~make-up~~ mode of operation accomplishes the actuation instrumentation's safety function.

B.1, B.2 and B.3

Condition B applies when the Required Action and associated Completion Time for Condition A have not been met. If movement of recently irradiated fuel assemblies is in progress, this activity must be suspended immediately to reduce the risk of accidents that would require CREFS actuation. In addition, if any unit is in MODE 1, 2, 3, or 4, the unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions for Condition B are modified by a Note that states that Required Action B.1 is not applicable for inoperability of the Containment Isolation actuation function. This note is necessary because the Applicability for the Containment Isolation actuation function is Modes 1, 2, 3, and 4. The Containment Isolation actuation function is not used for mitigation of accidents involving the movement of recently irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.5-1 determines which SRs apply to which CREFS Actuation Functions.

SR 3.3.5.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. However, in the case of the control room area and control room intake noble gas monitors, no independent instrument channel exist, therefore, the CHANNEL CHECK for these monitors will consist of a qualitative assessment of expected channel behavior, based on current plant and

BASES

APPLICABLE SAFETY ANALYSES (continued)

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from each steam generator (SG) is 500 gpd or increases to 500 gpd as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 2) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves. The 500 gpd primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 500 gpd primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR ~~100 or the staff approved licensing basis (i.e., a small fraction of these limits)~~50.67 (Ref. 5).

The RCS operational LEAKAGE satisfies Criterion 2 of ~~the NRC Policy Statement~~10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 4. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows. The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 4).

REFERENCES

1. FSAR Section 1.3.3.
2. FSAR, Section 14.
3. NEI 97-06, "Steam Generator Program Guidelines."
4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."
5. 10 CFR 50.67

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The ~~maximum dose to the whole body and the thyroid~~ total effective dose equivalent (TEDE) that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR ~~100~~ 50.67 (Ref. 1). The limits on specific activity ensure that the doses are held to ~~a small fraction of~~ the 10 CFR ~~100~~ 50.67 limits during analyzed transients and accidents as described in Table 6 of Regulatory Guide 1.183 (Ref. 3).

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to ~~a small fraction of~~ the 10 CFR ~~100~~ 50.67 dose guideline limits as described in Table 6 of Regulatory Guide 1.183. ~~The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations. The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits.~~ The values were based on a value of 0.5 µCi/g. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed ~~a small fraction of~~ the 10 CFR ~~100~~ 50.67 dose guideline limits as described in Table 6 of Regulatory Guide 1.183 following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of ~~1~~ 0.35 gpm per steam generator. The safety analysis assumes the specific activity of the secondary coolant at its limit of ~~1.0~~ 0.1 µCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.13, "Secondary Specific Activity."

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

~~The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 0.8 0.5 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 50 335 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 50 60 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of 100/Ē μCi/gm for gross specific activity.~~

The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the atmospheric steam dump valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of an SGTR accident ~~are within a small fraction of~~ will meet the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to ~~50~~ 60 μCi/gm DOSE EQUIVALENT I-131.

BASES

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR ~~100~~ 50.67 dose guideline limits.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10 CFR 50.36(c)(2)ii).

LCO

The specific iodine activity is limited to ~~0.8~~ 0.5 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the ~~2 hour thyroid dose~~ TEDE calculated for ~~to~~ an individual at the site boundary during the Design Basis Accident (DBA) will ~~be a small fraction of~~ meet the allowed TEDE acceptance criteria. ~~thyroid dose~~. ~~The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.~~

The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR ~~100~~ 50.67 dose guideline limits.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR ~~100.11, 1973~~ 50.67.
2. FSAR, Section 14.2.4.
3. Regulatory Guide 1.183, July 2000.

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate greater than or equal to equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via safety valves.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 500 gallons per day or is assumed to increase to 500 gallons per day as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR ~~100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits)~~50.67 (Ref. 3) dose guideline limits as described in Table 6 of Regulatory Guide 1.183..

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, an SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, "Steam Generator Program," and describe acceptable SG tube performance.

BASES

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR ~~100~~ 50.67.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

BASES

BACKGROUND (continued)

exchangers, and the SI pumps. Each of the two subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. ECCS Train interconnections could allow utilization of components from the opposite ECCS train to achieve the required ECCS flowpaths; however, cross train operation in the recirculation mode of operation requires local valve manipulations. Based on estimated times to establish the required valve line ups, the capability of establishing ECCS recirculation mode without interrupting injection flow to the core could be impaired. Therefore, with more than one component inoperable such that both Trains of ECCS are inoperable, the facility is in a condition outside of its design basis.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the steam generators provide core cooling until the RCS pressure decreases below the SI pump shutoff head.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment sump. The RHR pumps then supply the SI pumps and the containment spray pumps.

The SI subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

The ECCS subsystems are actuated upon receipt of an SI signal. If offsite power is available, the safeguard loads start immediately. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, upper plenum injection line valve stroke, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet the Point Beach Design Criteria (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting design basis Loss of Coolant Accident without exceeding the design leakage rate.

For the design basis Loss of Coolant Accident analyses, it is assumed that the containment is OPERABLE such that, the release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of ~~0.4~~ 0.2% of containment air weight per day (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be ~~0.4~~ 0.2% per day in the safety analysis at P_a = 60 psig (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits contained in the Containment Leakage Rate Testing Program must be met.

Compliance with this LCO will ensure a containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of $1.0\ L_a$.

BASES

APPLICABLE SAFETY ANALYSES (continued)

This peak pressure is less than the containment design pressure of 60 psig.

The analysis of the Main Steam Line Break (MSLB) offsite radiological consequences uses the analytical methods and assumptions outlined in ~~the Standard Review Plan~~ Regulatory Guide 1.183 (Ref.~~erence~~ 5). For the pre-accident iodine spike, it is assumed that a reactor transient has occurred prior to the MSLB and has raised the RCS iodine concentration to ~~the allowed Technical Specification~~ a conservative value of ~~50~~ 60 μCi/gm of dose equivalent (DE) I-131 at 100% power. For the accident-initiated iodine spike, the reactor trip associated with the MSLB creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value of 500 times greater than the release rate corresponding to the maximum equilibrium RCS Technical Specification concentration of ~~0.8~~0.5 μCi/gm of DE I-131. The affected SG will rapidly depressurize and release to the outside atmosphere the radioiodines initially contained in the secondary coolant and the radioiodines which are transferred from the primary coolant through SG tube leakage. A portion of the iodine activity initially contained in the intact SGs and noble gas activity due to tube leakage is released to atmosphere as well. The amount of primary to secondary SG tube leakage in each of the two SGs is assumed to be ~~equal to the Technical Specification limit for a single SG of~~ 0.35 gpm per steam generator. 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 ruptured main stream line is assumed to boil dry. 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 SG tube leaks is assumed to be released directly to the environment with no credit taken for iodine retention in the SG.

Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generator to

BASES

APPLICABLE SAFETY ANALYSES (continued)

minimize radiological releases.

In addition to providing SG isolation during a SLB or SGTR, the MSIVs are also containment isolation valves. The containment isolation function of these valves is addressed under LCO 3.6.3.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that two MSIVs and two non-return check valves in the steam lines are to be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal. The steam line non-return check valves are considered to be operable when they are capable of closing in response to reverse flow.

This LCO provides assurance that the MSIVs and non-return check valves will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR ~~100~~ 50.67 (Ref. 3).

APPLICABILITY

The MSIVs and non-return check valves must be OPERABLE in MODES 1, 2, and 3, when there is significant mass and energy in the RCS and steam generators.

In MODE 4, normally the MSIVs and non-return check valves are closed, and the steam generator energy is low.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs and non-return check valves are not required for isolation of potential high energy secondary system pipe breaks in these MODES .

ACTIONS

A.1

With one or more valves in a SG flowpath inoperable in MODE 1, action must be taken to restore the flowpath to OPERABLE status within 8 hours. Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs or non-return check valves.

The MSIVs are containment isolation valves, and as such the applicable Conditions and Required Actions of LCO 3.6.3 must be

BASES

SURVEILLANCE REQUIREMENTS (continued)

establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies that each MSIV will actuate to its isolation position on a actuation isolation signal. The 18 month Frequency is based on a refueling cycle interval and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components normally pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that allows entry into and operation in MODES 2 and 3 prior to performing the SR. This allows delaying testing until conditions where the testing can be performed are established.

SR 3.7.2.3

This SR verifies that each main steam non-return check valve can close. As the non-return check valves are not tested at power, they are exempt from the ASME Code (Ref. 4) requirements during operation in MODE 1, 2, or 3. The Frequency is in accordance with the Inservice Testing Program. Operating experience has shown that these components usually pass the Surveillance when performed at the Frequency required by the Inservice Testing Program. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 10.1.
2. FSAR, Section 14.2.5.
3. 10 CFR ~~100.115~~ 50.67.
4. ASME OM Code, Code for Operation and Maintenance of Nuclear Power Plants.
5. ~~NUREG-0800, Standard Review Plan 15.1.5, Appendix A, "Radiological Consequence of Main Steam Line Failures Outside of a PWR", Rev. 2, July 1981.~~ Regulatory Guide 1.183, July 2000.

B 3.7 PLANT SYSTEMS

B 3.7.9 Control Room Emergency Filtration System (CREFS)

BASES

BACKGROUND

The CREFS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity.

The CREFS consists of one ~~emergency~~ make-up air filtration unit, two ~~emergency~~ make-up fans, two recirculation fans, and the required ducts and dampers necessary to establish the required flow paths and isolation boundaries. The CREFS is an emergency system, parts of which operate during normal unit operations. The CREFS has ~~four~~ five modes of operation.

- Mode 1 (normal operation) - One of the two recirculation fans (W-13B1 or W-13B2) are in operation. Outside air is supplied from an intake penthouse located on the roof of the auxiliary building at a rate of approximately 1000 cfm (5% of system design flow) via damper VNCR-4849C which is throttled to a predetermined position. The make-up air combines with return air from the control room and computer room then passing through filter (F-43) and cooling units (HX-100 A&B) before entering the recirculation fan. Filtered and cooled air is supplied to the mechanical equipment room and through separate heating coils (HX-92 and HX-91 A&B), and humidifiers (Z-78 and Z-77) to the computer and control rooms respectively. Room thermostats and humidistats control the operation of the heating coils, chilled water system, and humidifiers. The control room heating, cooling, and humidification systems are not required to demonstrate compliance with the control room habitability limits of 10 CFR 50 Appendix A, GDC-19 as required by NUREG-0737, Item III.D.3.4. The computer room is supplied with supplementary cooling during normal operation via supplementary air conditioning units (W-107A/HX-190A/HX-191A or W-107B/HX-190B/HX-191B). Nominally, the control room washroom exhaust fan (W-15) is also in operation. Operation of the Control Room Ventilation System in mode 1 (normal operation) is not assumed for control room habitability, and is therefore not a Technical Specification required mode of operation.

BASES

BACKGROUND (continued)

- Mode 2 (recirculation operation) - 100% of the control room and computer room air is recirculated. In this mode, the outside air damper (VNCR-4849C) is closed and the control room washroom exhaust fan is de-energized. Recirculation can be automatically initiated by a Containment Isolation or Safety Injection signal, or can be manually initiated from the control room. Operation of the Control Room Ventilation System in mode 2 (recirculation) is not assumed for control room habitability, and is therefore not a Technical Specification required mode of operation.

- Mode 3 (recirculation/charcoal adsorber operation) - One of two control room ~~emergency~~ make-up fans (W-14A or W-14B) is in operation and air is supplied to the ~~emergency~~ make-up charcoal filter unit (F-16) via the computer and control room return air duct (damper VNCR-4851B). The normal outside air supply is secured (damper VNCR-4849C closed) and the control room washroom exhaust fan is de-energized. In this mode approximately 25% of the return air is being recirculated by the ~~emergency~~ make-up charcoal filter unit back to the suction of the control room recirculation fans. Recirculation/charcoal adsorber mode is manually initiated from the control room. Operation of the Control Room Ventilation System in mode 3 (recirculation/charcoal adsorber mode) is not assumed for control room habitability, and is therefore not a Technical Specification required mode of operation.

- Mode 4 (~~emergency make-up~~ HEPA/charcoal filtered outside air makeup) - Operation in this mode is similar to mode 3 except return air inlet damper VNCR-4851B to the ~~emergency~~ make-up fans remains closed and outside air supply to the ~~emergency~~ make-up charcoal filter unit opens (damper VNCR-4851A). This allows approximately 4950 cfm (25% of system design flow) of make-up air to pass through the emergency make-up charcoal filter unit to the suction of the control room recirculation fan. This make-up flow rate is sufficient to assure a positive pressure ~~of ≥ 1/8 in. water gage~~ is maintained in the control and computer rooms to limit excessive unfiltered in-leakage into the control room ventilation boundary. ~~Mode 4 (emergency make-up) is automatically initiated by a high radiation signal from the control room area monitor RE-101, or a high radiation signal from noble gas monitor RE-235 located in the supply duct to the control room.~~ This mode of operation ~~can also be~~ is manually initiated from the control room. Operation of the Control Room Ventilation System in Mode 4 (HEPA/charcoal filtered outside air makeup) is not assumed for control room habitability, and is therefore, not a Technical Specification required mode of operation. ~~Operation of the Control Room Ventilation System in mode 4 (emergency make-up) is the assumed mode of operation for the control room habitability analysis, and is therefore the only mode of operation addressed by this LCO.~~

BASES

BACKGROUND (continued)

- Mode 5 (emergency HEPA/charcoal filtered outside air and HEPA/charcoal filtered return air) operation is similar to mode 4 except that the return air inlet damper VNCR-4851B to the emergency fans opens. This allows a combination of outside air (<2500 cfm) and return air to pass through the emergency HEPA/charcoal filter unit to the suction of the control room recirculation fan for a total flow rate of 4950 cfm ±10%. This makeup flow rate is sufficient to assure a positive pressure that will prevent excessive unfiltered in-leakage into the control room ventilation boundary. Mode 5 is automatically initiated by a containment isolation signal, or by a high radiation signal from the control room monitor RE-101, or by a high radiation signal from the noble gas monitor RE-235 located in the supply duct to the control room. This mode of operation can also be manually initiated from the control room. Operation of the CREFS in Mode 5 is the assumed mode of operation for the control room habitability analyses, and is therefore, the only mode of operation addressed by this LCO.

~~The air entering the control room is continuously monitored by noble gas radiation monitors and the control room itself is continuously monitored by an area radiation monitor. One detector output above its setpoint will actuate the emergency make-up mode of operation (mode 4) for the CREFS.~~

BASES

BACKGROUND (continued)

The limiting design basis accident for the control room dose analysis is the large break LOCA. CREFS does not automatically restart after being load shed following a loss of offsite power; manual action is required to restart CREFS. Although it has been demonstrated that a loss of offsite power does not need to be assumed coincident with a LOCA with respect to CREFS system analysis and control room habitability, the control room emergency make-up and recirculation fans have been included in the emergency diesel generator loading profile during the recirculation phase of a loss of coolant accident.

The CREFS will pressurize the control and computer rooms ~~to at least 0.125 inches water gauge~~ in the emergency ~~make-up~~ mode of operation. The CREFS role in maintaining the control room habitable is discussed in the FSAR, Section 9.8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CREFS provides airborne radiological protection for control room personnel, as demonstrated by the limiting control room dose analyses for the design basis large break loss of coolant accident. Control room dose analysis assumptions are presented in the FSAR, Section 14.3.5 (Ref. 2).

The CREFS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The CREFS (mode ~~4~~5) is required to be OPERABLE to ensure that the control room habitability limits are met following a limiting design basis LOCA. Total system failure could result in exceeding the control room operator ~~thyroid dose limit of 30 rem~~ total effective dose equivalent (TEDE) limit of 5 rem in the event of a large radioactive release. The CREFS is considered OPERABLE when the individual components necessary to filter and limit control room in-leakage are OPERABLE. CREFS is considered OPERABLE when:

a. Both emergency make-up fans (W-14A and W-14B) are OPERABLE;

b. One recirculation fan (W-13B1 or W-13B2) is OPERABLE;

c. Emergency make-up filter unit (F-16), HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions;

d. Control room ventilation envelope is capable of achieving and maintaining a positive pressure ~~of at least 0.125 inches water gauge~~ in the emergency ~~make-up~~ mode of operation;

BASES

LCO (continued)

e. Ductwork and dampers are OPERABLE, and air circulation can be maintained; and

f. CREFS is capable of being manually initiated in the emergency ~~make-up~~ mode of operation (mode ~~4~~5).

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

APPLICABILITY

In MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies, CREFS must be OPERABLE to control operator exposure during and following a DBA.

During movement of irradiated fuel assemblies, the CREFS must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

A.1

When CREFS is inoperable, action must be taken to restore the system to OPERABLE status within 7 days. The 7 day Completion Time is based on the low probability of a DBA challenging control room habitability occurring during this time period.

B.1, B.2 and B.3

If CREFS cannot be restored to OPERABLE status within the required Completion Time with movement of irradiated fuel in progress, this activity must be suspended immediately. Immediately suspending this activity places the unit in a condition that minimizes risk from this activity. This does not preclude the movement of fuel to a safe position.

In MODE 1, 2, 3, or 4, if CREFS cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS (continued)

designed to pressurize the control room ~~≥ 0.125 inches water gauge~~ to a positive pressure with respect to adjacent areas in order to minimize unfiltered inleakage. The CREFS is designed to maintain ~~this~~ a positive pressure with one emergency make-up fan in operation at a makeup flow rate of ±10% of the nominal make-up pressurization flow rate of approximately 4950 cfm. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800 (Ref. 4).

REFERENCES

1. FSAR. Section 9.8.
2. FSAR. Section 14.3.5.
3. Regulatory Guide 1.52, Rev. 2.
4. NUREG-0800, Section 6.4, Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.13 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

The release of secondary system activity is assumed in several accidents to include reactor coolant pump locked rotor, control rod ejection, steam generator tube rupture, and Main Steam Line Break. The MSLB is the most limiting relative to secondary activity and is therefore used to establish the secondary coolant activity limit.

The MSLB involves a complete severance of a main steam line outside containment. The affected SG will rapidly depressurize and release to the outside atmosphere all of the radioiodines initially contained in the SG and the radioiodines which are transferred from the primary coolant through SG tube leakage. Iodine and noble gas activity is also released from the intact SG. A portion of the iodine activity initially contained in the intact SG is released, in addition to radioiodines and noble gases from the RCS through SG tube leakage, during plant cooldown to Residual Heat Removal entry conditions.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR ~~100~~ 50.67 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter 14.2.5 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of ~~1.~~0.1 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the MSLB. The MSLB offsite radiological analysis uses the analytical methods and assumptions outlined in the Standard Review Plan (Ref. 3). The result of the radiological analysis for this

BASES

APPLICABLE SAFETY ANALYSES (continued)

event shows that the radiological consequences of an MSLB do not exceed ~~a small fraction of the plant Exclusion Area Boundary limits (Ref. 1) for whole body and thyroid dose rates~~ the limits specified in 10 CFR 50.67 (Ref. 1).

~~Two offsite dose analyses are performed, one assuming a pre-accident RCS iodine spike, and the second involving an RCS iodine spike as a result of the MSLB. For the pre-accident iodine spike, it is assumed that a reactor transient has occurred prior to the MSLB which has raised the RCS DOSE EQUIVALENT I-131 concentration to the allowed Technical Specification value of 50 μCi/gm. For the accident-initiated iodine spike, the reactor trip associated with the MSLB creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value of 500 times greater than the release rate corresponding to the maximum proposed equilibrium RCS DOSE EQUIVALENT I-131 Technical Specification concentration of 0.8 μCi/gm. The duration of the accident-initiated iodine spike is assumed to be 1.6 hours.~~

~~The following is a summary of other major assumptions and parameters used in both the pre and post accident cases outlined above:~~

1. ~~Primary and secondary system activities are at equilibrium prior to the accidents.~~

2. ~~The RCS noble gas activity is based on a fuel defect level of 1.0%. This is approximately equal to 100/E-bar μCi/gm for gross radioactivity.~~

3. ~~The secondary coolant iodine activity is assumed to be 1.0 μCi/gm of DOSE EQUIVALENT I-131.~~

4. ~~Primary to secondary SG tube leakage in each SGs is assumed to be 0.35 gpm.~~

5. ~~The atmospheric dispersion factor (χ/Q) at site boundary during the two hours following the accident is 5.0×10^{-4} m^3/sec.~~

6. ~~Breathing rate used to calculate the thyroid dose for the accidents is 3.47×10^{-4} m^3/sec.~~

7. ~~The SG connected to the ruptured main steam line is assumed to boil dry within 30 minutes.~~

8. ~~All of the activity contained in the steam generator connected to the ruptured steam line is assumed to be released directly to the environment. No credit is taken for activity plate out or retention.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~9. Iodine carried over to the faulted SG by SG tube leaks is assumed to be released directly to the environment.~~

~~10. No credit is taken for iodine removal from steam released to the condenser prior to reactor trip and concurrent loss of offsite power.~~

~~11. With the loss of offsite power, the remaining intact steam generator is available for core decay heat removal by venting steam to the atmosphere.~~

~~12. The intact steam generator is assumed to discharge entrained activity to the atmosphere. The iodine partition factor for the intact SG is assumed to be 0.01.~~

~~13. The Auxiliary Feedwater System supplies makeup to the intact steam generator.~~

~~14. Venting of steam from the intact SG continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to be placed into operation to complete the cooldown. Eight hours after the accident, the residual heat removal system is assumed to be placed into operation.~~

With the loss of offsite power, the unfaulted steam generator is available for core decay heat dissipation by venting steam to the atmosphere through the main steam safety valves (MSSVs) or atmospheric dump valve (ADV). The auxiliary feedwater system supplies the necessary makeup to the unfaulted steam generator. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the residual heat removal system to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unfaulted steam generator is assumed to discharge steam and any entrained activity through the MSSVs or ADV during the event. The resultant radiological consequences represent a conservative estimate of the potential integrated doses due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of ~~the NRC Policy Statement.~~ 10 CFR 50.36(c)(2)(ii).

BASES

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ ~~1.0~~ 0.1 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) ~~to a small fraction of the required limit (Ref. 1).~~ as specified in 10 CFR 50.67 (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gross beta-gamma or gamma isotopic analysis of the secondary coolant, may be used to confirm DOSE EQUIVALENT I-131 is ≤ ~~1.0~~ 0.1 μCi/gm. Confirmation of gross activity is a conservative means of determining compliance with the LCO limit. However, if gross activity exceeds the 1.0 μCi/gm limit, an isotopic analysis should be performed to determine DOSE EQUIVALENT I-131, to prevent unnecessary shutdowns. Performance of this SR confirms the validity of the safety analysis assumptions as to the secondary system source terms for post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

BASES

REFERENCES

1. 10 CFR ~~100.11~~ 50.67.
2. FSAR. Chapter 14.2.5.
3. NUREG 0800, USNRC Standard Review Plan, 15.1.5, Steam Piping Failures Inside and Outside of Containment (PWR), Rev. 2, July 1981.
4. Regulatory Guide 1.183, July 2000

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal.

The Surveillance is performed every 7 days during movement of recently irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide assurance that containment penetrations are in their required position during the applicable period for this LCO.

SR 3.9.3.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

The SR is modified by a Note stating that this demonstration is not applicable to valves in isolated penetrations. LCO 3.9.3.c.1 provides the option to close penetrations in lieu of requiring automatic isolation capability.

REFERENCES

1. FSAR. Section 14.2.1

2. NUREG-0800, Section 15.7.5, Rev. 1, July 1981

3. 10 CFR 50.67.

4. Regulatory Guide 1.183 ~~(Rev 0)~~ July 2000.

ENCLOSURE 3

LICENSE AMENDMENT REQUEST 241
TECHNICAL EVALUATION

POINT BEACH NUCLEAR PLANT

72 Pages Follow

1.0 Introduction

1.1 Evaluation Overview and Objective

The objective of this technical report is to document the Point Beach Nuclear Plant (PBNP) full implementation of the Alternative Source Terms (AST) in accordance with 10 CFR 50.67 (Reference 3) as described in Regulatory Guide (RG) 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 2). The offsite and control room dose analyses for the following Final Safety Analysis Report (FSAR) accidents have been reanalyzed to satisfy a regulatory commitment to revise the current accident analysis, demonstrating compliance with the dose limits of General Design Criteria (GDC)-19 using the Alternative Source Term (Reference 22). As part of this submittal, the post-accident reliance on potassium iodide (KI) for control room staff has been addressed.

FSAR Sections:

14.1.8, Loss of Coolant Flow (Locked Rotor - LR)
14.2.1, Fuel Handling Accident (FHA)
14.2.4, Steam Generator Tube Rupture (SGTR)
14.2.5, Rupture of a Steam Pipe (Main Steam Line Break - MSLB)
14.2.6, Rupture of a Control Rod Mechanism Housing – RCCA Ejection (CRDE)
14.3.5, Large Break Loss of Coolant Accident (LOCA)
14.3.6, Reactor Vessel Head Drop Accident (RVHD)

Each accident listed above, along with the specific input and assumptions, is described in Section 6.0 of this Enclosure 3.

The FHA, based on the AST, had been submitted previously under Reference 20 and approved under Reference 19. It is represented in this amendment to present the change in the dose to the control room due to a proposed control room heating, ventilation and air conditioning (HVAC) emergency mode change, change in the source term and a change in the atmospheric dispersion factors for the control room. An evaluation was performed to determine the impact on calculated dose results for the FHA where some nuclear fuel assemblies used in the PBNP FHA radiological analysis did not meet the Table 3 Footnote 11 criteria of RG 1.183. For this evaluation it was assumed that the fuel that did not meet the RG 1.183 Table 3 Footnote 11 limits had higher gap fractions. The higher gap fractions were applied only to the fraction of the fuel that did not meet the limit, following a method previously approved by the NRC for the Kewaunee Power Station (Reference 24). The gap fractions used were those from Safety Guide 25 with the value for I-131 adjusted consistent with the recommendation in NUREG/CR-5009. RG 1.183 Table 3 gap fractions used in the base analysis for fuel that met the Footnote 11 limits were then compared to the gap fractions that were applied for fuel that did not meet the Footnote 11 limits. Additionally, calculations were completed assuming all the fuel in the assembly did not meet Footnote 11 limits.

1.2 Changes to the PBNP Design and Licensing Basis

The following denotes the more significant proposed changes to the PBNP design and licensing bases:

1. The AST methodology is adopted for the composition, magnitude, chemical form, and timing of radiation releases, as well as accident specific modeling for all radiological design basis accidents (DBAs) presented in the PBNP FSAR;

2. The radiological acceptance criteria is changed from 10 CFR 100 (whole body/critical organ) to 10 CFR 50.67 total effective dose equivalent (TEDE);

3. Atmospheric dispersion factors for the control room intake are reanalyzed for existing pathways using ARCON96;

4. An increased value is assumed for unfiltered in-leakage to the control room;

5. Control room ventilation system operating with filtered recirculation in addition to filtered fresh air intake;

6. Credit is taken for future shielding modifications to the control room;

7. A reduced value in the allowable dose equivalent (DE) I-131 concentrations in the primary and secondary systems is used;

8. A reduced containment leakage is modeled;

9. Credit is taken for the use of containment spray while on emergency core cooling system (ECCS) recirculation LOCA;

10. A factor of two increase is applied to the ECCS leakage limit for control room habitability radiological analysis;

11. Flashing fractions are applied to the SGTR break flow;

12. Credit is not taken for the administration of KI to control room personnel;

13. Elimination of the requirement that the reactor must be shut down for 100 hours prior to lifting the reactor vessel head based on the revised analyses.

1.3 Regulatory Guide 1.183 Implementation Provisions

Except as noted, the revised PBNP accident analyses addressed in this submittal follow the guidance provided in Regulatory Guide 1.183:

1. Consideration of the loss of offsite power (LOOP) is taken in all accidents with regard to accident mitigation systems in order to maximize the release from a plant system. In general, the LOOP was used to limit equipment availability for plant cooldown, which in turn, results in a larger amount of activity being released. NRC has acknowledged that a

LOOP need not be considered coincident with a LOCA for the purposes of evaluating control room habitability. (Reference 5)

2. For accidents that assume a coincident iodine spike, the spike duration is based on the amount of activity available for release from the gap of fuel pins with defects. The guidance provided in RG 1.183 would generally assume an 8 hour spike but allows for shorter durations if it can be shown that there is not enough available activity for release for the entire duration.

1.4 Computer Codes

	Code	Application
1	Industry computer code, "ORIGEN2.1: Isotope Generation and Depletion Code –Matrix Exponential Method," RSIC Computer Code Collection	Used to determine the equilibrium core inventory. ORIGEN2 is a versatile point depletion and radioactive decay computer code for use in simulating nuclear fuel cycles and calculating the nuclide compositions and characteristics of materials contained therein.
2	Westinghouse Proprietary Computer Code, "FIPCO," Version 3.1	Used to calculate the reactor coolant system (RCS) maximum coolant activity obtained during a cycle of operation.
3	Westinghouse Proprietary Computer Program, "TITAN5," Version 4.10	Used to model releases and calculate offsite and control room doses.
4	S&W Computer Code, SW-QADCGGP, "A Combinatorial Geometry Version of QAD5A," NU-222, V00, L01	S&W version of industry standard point kernel program "QADCGGP" used to calculate control room operator dose by modeling source-shield-detector configurations.
5	S&W Proprietary Computer Code, PERC2, "Passive Evolutionary Regulatory Consequence Code," NU-226, V00, L01	Used to generate the source terms in the containment atmosphere, in the external plume passing the control room, and the control room charcoal and HEPA filters to determine control room operator dose.
6	Westinghouse Proprietary Computer Code, "CIRCUS," Version 1.1	Used to calculate the spray droplet average diameters and fall times for use in the calculation of elemental and particulate iodine removal coefficients.
7	NRC Sponsored ARCON96, "Atmospheric Relative Concentrations in Building Wakes," developed by Pacific Northwest Laboratory	Used to calculate atmospheric dispersion factors (χ/Q) for control room doses.

2.0 Radiological Evaluation

2.1 Introduction

The Point Beach Nuclear Plant (PBNP) licensing basis for the radiological consequences analyses currently utilizes methodologies, assumptions and dose limits that are derived from Technical Information Document (TID)-14844 "Calculation of Distance Factors for Power and Test Reactor Sites" (Reference 1) and other Standard Review Plan guidance. RG 1.183 (Reference 2) provides guidance for application of AST in revising the accident source terms used in design basis radiological consequences analyses, as allowed by 10 CFR 50.67 (Reference 3). The AST methodology as established in RG 1.183 is being used to calculate the offsite and control room radiological consequences for PBNP to support the increase in the assumed control room unfiltered inleakage value to address the reliance on KI. The following accidents are analyzed: Large Break Loss of Coolant Accident (LOCA), Steam Generator Tube Rupture (SGTR), Locked Rotor (LR), Main Steam Line Break (MSLB), Control Rod Ejection (CRDE) and Reactor Vessel Head Drop Accident (RVHD). Each accident and the specific input assumptions are described in detail in subsequent sections in this Enclosure.

The fuel handling accident has been previously analyzed using the AST methodology. NRC approval was given to PBNP via Safety Evaluation Report dated April 2, 2004 (Reference 19). This accident is represented in this technical evaluation for consistency, as well as, to provide the impact on the control room doses due to various input parameter changes.

The current licensed maximum reactor power level is 1540 MWt. The analyses in this Enclosure model a maximum power of 1683 MWt. Although the analyses were performed at a higher power level, this license amendment is not requesting approval for use of the higher power.

2.2 Common Analysis Inputs and Assumptions

The assumptions and inputs described in this section are common to analyses discussed in this Enclosure. The accident specific inputs and assumptions are discussed in Sections 6.1-6.7.

The TEDE doses are determined at the exclusion area boundary (EAB) for the worst 2 hour interval. The TEDE doses at the low population zone (LPZ) are determined for the duration of the event. The TEDE doses for the control room are calculated for 30 days. This duration typically extends beyond the time when the releases are terminated. This accounts for the additional dose to the operators in the control room, which will continue for as long as the activity is circulating within the control room envelope.

The TEDE dose is equivalent to the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. Effective dose equivalent (EDE) is used in lieu of DDE in determining the contribution of external dose to the TEDE dose consistent with RG 1.183 guidance. The dose conversion factors (DCFs) used in determining the CEDE dose are from EPA Federal Guidance Report No. 11 (Reference 6) and are given in Table 1. The dose conversion factors used in determining the EDE dose are from EPA Federal Guidance Report No. 12 (Reference 7) and are listed in Table 2.

The offsite breathing rates and the offsite atmospheric dispersion factors used in the offsite radiological calculations are provided in Table 3. The offsite dispersion factors used to assess offsite dose are those currently part of the PBNP licensing basis.

Parameters used in the control room personnel dose calculations are provided in Table 4. These parameters include the normal operation flow rates, the post-accident operation flow rates, control room volume, filter efficiencies and control room operator breathing rates. Atmospheric dispersion factors are specific to each analysis and are calculated with respect to the location of control room intake. These accident specific atmospheric dispersion factors are applied to the unfiltered inleakage value as well. The atmospheric dispersion factors are further discussed in Section 4.0 and are listed with the accident specific input parameter tables.

Each non-LOCA accident assumes an unfiltered inleakage value of 330 cfm into the control room. The LOCA accident assumes an unfiltered inleakage value of 105 cfm. Current licensing basis analysis for control room habitability assumes an unfiltered inleakage of 10 cfm based on the guidance of the Murphy-Campe methodology (Reference 9). This early methodology provided a value of 10 cfm unfiltered inleakage for pressurized control rooms, in order to account for the door opening/closing.

Recent industry inleakage testing of control room envelopes has shown that 10 cfm may not be a conservative value for this parameter (Reference 21). In light of this industry issue, PBNP has made improvements to the integrity of the envelope in order to minimize the potential for unfiltered inleakage as well as reduce overall operator dose post-accident. The control room envelope and habitability is discussed further in Section 5.0 of this enclosure.

No credit is taken for the radioactive decay during release and transport or for cloud depletion by ground deposition during transport to the control room, EAB or outer boundary of the LPZ. Decay is a depletion mechanism credited only for a source term prior to release to the atmosphere and for activity after it enters the control room. Decay constants for each nuclide are provided in Table 7.

The primary-to-secondary leakage assumed in the SG is applicable in all of the accidents except FHA, RVHD and LOCA. The amount of primary-to-secondary SG leakage is conservatively assumed to be 0.7 gpm total (i.e., Technical Specification (TS) 5.5.8 accident induced leakage performance criteria of 500 gpd per SG). The density for this leakage is assumed to be 47 lb_m/ft^3.

Surveillance tests and facility instrumentation used to show compliance with the leak rate TS are done under cooled liquid conditions. However, final calculations take into consideration the nominal temperature and pressure of the RCS and secondary side of the SG. The plant procedures assume the average RCS fluid temperature, T_{avg}, whereas the analyses assumed the lowest temperature in the allowable range yielding a slightly larger density. Although the primary-to-secondary pressure differential drops throughout these events, the constant flow rate is conservatively maintained.

The core fission product activity is provided in Table 5 for all nuclides that are addressed. The nominal reactor coolant activity based on 1 percent fuel defects that corresponds to the TS 3.4.16 limit of 100/E-bar is provided in Table 6. The core and coolant activities in Tables 5 and 6 are based on a core power of 1650 MWt increased to 1683 MWt to include a 2 percent power uncertainty. The core fission product activity and nominal reactor coolant activity are discussed in detail in Section 3.0.

2.3 Dose Calculation Models

Offsite Dose Calculation Models

The TEDE dose is calculated for the worst 2 hour period at the EAB. At the LPZ, the TEDE dose is calculated up to the time all releases are terminated. The TEDE doses are obtained by combining the CEDE doses and the EDE doses.

Offsite inhalation doses (CEDE) are calculated using the following equation:

$$D_{CEDE} = \sum_i \left[DCF_i \left(\sum_j (IAR)_{ij} (BR)_j (\chi/Q)_j \right) \right]$$

Where:

D_{CEDE} = CEDE dose via inhalation (rem).
DCF_i = CEDE dose conversion factor via inhalation for isotope i (rem/Ci) (Table 1)
$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j (Ci)
$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table 3)
$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3) (Table 3)

Offsite external exposure (EDE) doses are calculated using the following equation:

$$D_{EDE} = \sum_i \left[DCF_i \left(\sum_j (IAR)_{ij} (\chi/Q)_j \right) \right]$$

Where:

D_{EDE} = external exposure dose via cloud submersion (rem)
DCF_i = EDE dose conversion factor via external exposure for isotope i (rem·m^3/Ci·sec) (Table 2)
$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j (Ci)
$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3) (Table 3)

Control Room Dose Calculation Models

CEDE (doses due to inhalation) and EDE (doses due to external exposure) are calculated for 30 days in the control room. The TEDE dose for the control room operator is calculated by adding the EDE dose to the CEDE dose.

The control room is modeled as a discrete volume. The atmospheric dispersion factors calculated for the transfer of activity to the control room intake are used to determine the activity available at the control room intake. The inflow (filtered and unfiltered) to the control room is used to calculate the concentration of activity in the control room. Control room parameters used in the analyses are presented in Table 4. Control room atmospheric dispersion factors used in each analysis are provided in the input assumption table for that accident (e. g. LOCA – Table 18).

Control room inhalation doses are calculated using the following equation:

$$D_{CEDE} = \sum_i \left[DCF_i \left(\sum_j Conc_{ij} * (BR)_j * (OF)_j \right) \right]$$

Where:

D_{CEDE} = CEDE dose via inhalation (rem)
DCF_I = CEDE dose conversion factor via inhalation for isotope i (rem/Ci) (Table 1)
$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered inflow, total outflow and CR volume (Ci-sec/m^3)
$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table 4)
$(OF)_j$ = occupancy factor during time interval j (Table 4)

Control room external exposure doses due to activity in the control room volume are calculated using the following equation:

$$D_{EDE} = \left(\frac{1}{GF} \right) * \sum_i DCF_i \left(\sum_j Conc_{ij} * (OF)_j \right)$$

Where:

D_{EDE} = external exposure dose via cloud submersion in rem.
GF = geometry factor, calculated based on Reference 9, using the equation:
$GF = \frac{1173}{V^{0.338}}$, where V is the control room volume in ft^3
DCF_i = EDE dose conversion factor via external exposure for isotope i (rem·m^3/Ci·sec) (Table 2)
$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered inflow, total outflow and CR volume (Ci-sec/m^3)
$(OF)_j$ = occupancy factor during time interval j (Table 4)

3.0 Radiation Source Terms

3.1 Core Inventory

A new core source term has been calculated for use in the radiological accident analyses. The inventory of the fission products in the reactor core is based on maximum full-power operation of the core at a power level equal to 1683 MWt, and current licensed values of fuel enrichment and burnup. The core mass calculated is 48.0 MTU with an equilibrium cycle length of 17,175 MWD/MTU. The fuel was modeled with an active fuel length of 132 inches with axial blanket regions of six inches in length. The current licensed core power level is 1540 MWt. Although the analyses were performed at the higher power level, this amendment request is not requesting approval for operation at the higher power level.

The ORIGEN2 computer code was used to determine the equilibrium core inventory. ORIGEN2 is a versatile point depletion and radioactive decay computer code for use in simulating nuclear fuel cycles and calculating the nuclide compositions and characteristics of materials contained therein. The equilibrium core inventory model runs a single assembly in each of seven regions. Sixteen new fuel assemblies with an active fuel enrichment of 4.40 w/o and 24 assemblies with an active fuel enrichment of 4.95 w/o are assumed. In each assembly, the axial blanket region is assumed to have an enrichment of 2.60 w/o. An average enrichment is calculated for each type of assembly for input to ORIGEN2.

Burnup calculations reflecting each of the appropriate power histories are performed. The ORIGEN2 model runs a single assembly in each of the seven regions; 16 assemblies of fresh (4.40 w/o), once-burnt and twice-burnt "A" regions, 24 assemblies of fresh (4.95 w/o), once burnt and twice burnt "B" regions, and a seventh region consisting of a single thrice-burnt "A" assembly. The total inventory for each region at the end of the equilibrium cycle is then determined by multiplying the assembly value by the number of assemblies per region. Finally, the seven regions are summed to produce a total core inventory. No shutdowns are modeled between cycles. While strictly conservative, this simplification is expected to have virtually no effect on core inventory. The equilibrium core at the end of a fuel cycle is assumed to consist of fuel assemblies with once, twice and thrice burnups.

The core inventory developed using ORIGEN2 based on the above methodology includes many isotopes that are not dose significant. Only those dose significant isotopes relative to light water reactor accidents are presented in Table 5.

3.2 Coolant Inventory

For the reactor coolant system, maximum coolant activities obtained during a cycle of operation are calculated. Small cladding defects in fuel are assumed present at initial core loading and uniformly distributed throughout the core. The radiation source is based on 1% of the rods containing fuel defects operating at the core power of 1683 MWt. The RCS activity is determined using a calculated minimum liquid volume to obtain conservative concentrations. Other parameters used in determining coolant inventory include the expected coolant cleanup flow rate, demineralizer effectiveness and volume control tank noble gas stripping behavior.

The chemical and volume control system (CVCS) is assumed to be operated with 40 gpm letdown flow and no purging of the volume control tank (VCT) in order to conservatively increase the RCS activities. By not purging, gases are retained in the vapor portion of the VCT thereby limiting any additional gases from exiting the reactor coolant. The source term for the RCS includes the decay products of the parent and daughter nuclides and assumes that there is no RCS leakage.

The Westinghouse QA Category I Proprietary computer code, FIPCO, is used to calculate the reactor coolant system (RCS) maximum coolant activity obtained during a cycle of operation. The core inventory generated by ORIGEN2, as discussed above, is relayed to FIPCO to determine the activity inventories in the RCS. The coolant activities tabulated are the maximum concentrations that occur during the fuel cycle from startup through the equilibrium cycle. Nuclides of fission and corrosion products listed in ANS Standard ANSI/ANS 18.1 (Reference 8) are included, as well as other nuclides important to dose and shielding calculations.

The reactor coolant activity inventory based on 1% fuel defect is listed in Table 6. The noble gas activity corresponds to TS 3.4.16 limit of 100/E-bar. The iodine activities presented in Table 6 have not been normalized to the proposed DE I-131 TS limit (TS 3.4.16) of 0.5 µCi/gm. In addition, the DE I-131 concentrations for the primary and secondary limits are listed in Table 21.

3.3 Gap Inventory for Non-LOCA Accidents

The gap fractions listed in Section 3.2 of RG 1.183 serve as a basis for determining available activity in the CRDE, MSLB, SGTR, RVHD, FHA, and LR radiological analyses. RG 1.183 Table 3 Footnote 11 states that the gap fractions are acceptable for Light Water Reactor fuel with a peak rod burnup less than 62,000 MWD/MTU provided the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54,000 MWD/MTU.

The CRDE uses the gap fraction for alkali metal from Table 3 of RG 1.183 and the fractions provided in Footnote 11 to RG 1.183 Table 3 for the iodines and nobles gases. The CRDE dose analysis is not dependent on the fuel meeting the RG 1.183 Table 3 Footnote 11 limitations on maximum linear heat generation rate and burnup.

The reactor coolant activity concentrations developed for MSLB and SGTR considered only iodines, noble gases and alkali metals to be consistent with Table 3 of RG 1.183. Additionally, the fraction of 8% I-131 from Table 3 of RG 1.183 serves as a basis for the iodine spike durations for MSLB and SGTR. Sections 6.2 and 6.4 provide additional discussion regarding iodine spiking.

The RVHD dose analysis modeled a 100% core gap release and applied the RG 1.183 Table 2 gap fractions and is not dependant on the fuel meeting the RG 1.183 Footnote 11 limitations on maximum linear heat generation rate and burnup.

An evaluation was performed to determine the impact on calculated dose results for the FHA where some nuclear fuel assemblies used in the PBNP FHA radiological analysis did not meet the Table 3 Footnote 11 criteria of RG 1.183. For this evaluation it was assumed that the fuel that did not meet the RG 1.183 Table 3 Footnote 11 limits had higher gap fractions. The higher gap fractions were applied only to the fraction of the fuel that did not meet the limit, following a method previously approved by the NRC for Kewaunee Power Station (ML070430020). The gap fractions used were those from Safety Guide 25 with the value for I-131 adjusted consistent with the recommendation in NUREG/CR-5009. RG 1.183 Table 3 gap fractions used in the base analysis for fuel that met the Footnote 11 limits were then compared to the gap fractions that were applied for fuel that did not meet the Footnote 11 limits:

FHA	RG 1.183 Footnote 11 Compliant Gap Fraction	Gap Fraction for Fuel not Meeting RG 1.183 Footnote 11
I-131	0.08	0.12
KR-85	0.1	0.3
Other iodines & noble gases	0.05	0.1

The result of the calculations determined that in the event of a fuel handling accident, the dose limits for the EAB, LPZ and Control Room are met even if none of the fuel in the failed assembly met the RG 1.183 Footnote 11 limits. The dose results for both cases are provided in Section 6.6 of this report.

The LR dose analysis was modeled using the gap fractions from Table 3 of RG 1.183, even though PBNP does not meet the limitations on maximum linear heat generation rate and burnup cited in Footnote 11 of RG 1.183.

The current Unit 1 Cycle 31 core contains 16 affected fuel assemblies. Of the 16 affected assemblies, the Westinghouse core model shows 78% of rods in the worst case assembly exceed the RG 1.183 Table 3 Footnote 11 assumption. In total, 1891 rods, or 8.73% of fuel rods in the core, exceed the RG 1.183 Table 3 Footnote 11 assumptions. These assemblies are all from fuel Region 31 (fuel assembly identifications beginning with HH), which had an initial enrichment of 4.95 w/o. The maximum enthalpy rise hot channel factor ($F\Delta H$) was determined to be 1.185. The end-of-life (EOL) peak rod burnup range for these assemblies is 60,938 MWD/MTU to 61,691 MWD/MTU. The linear heat generation rate is projected to be 6.87 kW/ft at EOL.

The current Unit 2 Cycle 29 core contains 12 affected fuel assemblies. Of the 12 affected assemblies, the Westinghouse core model shows 6.7% of rods in the worst case assembly exceed the RG 1.183 Table 3 Footnote 11 assumption. In total, 95 rods, or 0.44% of fuel rods in the core, exceed the RG 1.183 Table 3 Footnote 11 assumption. These assemblies are all from fuel Region 29B (fuel assembly identifications beginning with FF), which had an initial enrichment of 4.50 w/o. The maximum $F\Delta H$ of the affected assemblies was determined to be 1.135. The much smaller percentage of affected rods is attributed to the lesser fuel enrichment and lesser power. The EOL peak rod burnup range for these assemblies is 56,948 MWD/MTU to 58,878 MWD/MTU. The linear heat generation rate is projected to be 6.48 kW/ft at EOL.

This condition was evaluated and it was determined acceptable due to the significant additional conservatisms applied in the LR radiological analysis where it is assumed that 100% of the core gap activity is released, along with the conservative gap fraction values of Table 3 of RG 1.183. RG 1.183 Table 3 gap fractions are provided for non-LOCA events which are generally assumed to result in only a fraction of the core being damaged. When only a portion of the core is being considered, the application of additional conservatism to the gap fractions to address high burnup fuel has been used.

Use of RG 1.183 Table 3 gap fractions for the LR does not represent a commitment to use RG 1.183 Table 3 gap fractions in future analyses. For the PBNP LR radiological analysis, the use of Table 3 gap fractions is a conservative assumption held over from analysis assumptions more consistent with limited fuel failure, rather than an added conservatism to address high burnup fuel. For an event with a full core failure the gap fractions provided in Table 2 of RG 1.183 could be applied instead of Table 3. Although provided for LOCA releases to containment, Table 2 provides acceptable core average gap fractions and has no restriction based on heat generation rate and burnup.

4.0 Accident Atmospheric Dispersion Factors (X/Q)

4.1 Meteorological Monitoring Program

The PBNP Meteorological Monitoring System consists of three towers. Two towers are located near shore and the third is located about 8 miles inland. The towers are separated from nearby obstructions by distances equal to at least 10 times the obstruction height to minimize disturbances in the wind field being measured. All instrument booms extend at least two tower widths from the tower and are oriented into the predominant wind direction. Temperature sensor aspirator shields are pointed horizontally, to the north, to minimize the tower's effect on measurements and the effect of solar radiation on the sensor.

A significant meteorological phenomenon affecting areas bordering a large body of water is the lake breeze. This phenomenon can result in the formation of a thermal internal boundary layer which can adversely affect the dispersion of atmospheric contaminants under certain conditions. The effect of Lake Michigan upon meteorology in the vicinity of PBNP was a major consideration in sitting the individual monitoring towers.

The primary meteorological tower is the southern tower located approximately 850 meters south-southeast of the protected area and about 40 meters inland of the Lake Michigan shoreline. The primary monitoring tower consists of a 45 meter tower instrumented with equipment at the 10 and 45 meter levels. This tower's location is such that it should be in the same meteorological regime as the plant with respect to localized lake effects.

The backup monitoring tower is installed approximately 500 meters northwest of the plant and approximately 300 meters inland of the Lake Michigan shoreline. This tower is instrumented at the 10 meter level to provide backup information in the event of a failure at the primary tower. The backup tower site was chosen so that it would usually be in the same meteorological regime as the plant, with respect to localized lake effects.

The inland tower is located about eight miles inland from PBNP. This tower is designed to provide information on the penetration of lake breezes inland from the shoreline.

In order to ensure the accuracy of the monitoring system, the metrological monitoring instruments are calibrated on a semi-annual schedule. Calibrations are also performed after major equipment malfunctions, equipment modifications and equipment replacements.

In addition to the monitored meteorological data, status and alarm information are transmitted to the plant control room. If the output from a parameter exceeds the operating range for that parameter, an error alarm is generated. Visual field site inspections are performed at each monitoring site on at least a monthly basis. The inspections check the physical integrity of the site, appearance of the sensors for any obvious signs of weather damage or faulty operation and verify that the signal conditioning equipment is operating properly.

4.2 Offsite Atmospheric Dispersion Factors

The atmospheric dispersion (χ/Q) values for the PBNP exclusion area boundary (EAB) and the low population zone (LPZ) are those from the current licensing basis. These values were developed from the guidance provided in Regulatory Guide 1.145 (Reference 23) and meteorological data collected at the site's primary tower from January 1, 1991, through

December 31, 1993. The offsite χ/Q values are presented in Table 3 and represent the maximum sector χ/Q values.

4.3 Control Room Atmospheric Dispersion Factors

The control room intake χ/Q values are calculated using the latest version of the "ARCON96: Atmospheric Relative Concentrations in Building Wakes" methodology (Reference 4). Input data consists of hourly on-site meteorological data, release characteristic such as release height, the building area affecting the release; and various receptor parameters such as its distance and direction from the release to the control room air intake and intake height.

The χ/Q values are based on metrological data collected at PBNP from September 2000 to September 2005. Each hour of data, at a minimum, has a validated wind speed and direction at the 10 meter and 45 meter levels and a temperature difference between the 45 and 10 meter levels.

All releases are conservatively treated as ground level as there are no releases at this site that are high enough to escape the aerodynamic effects of the plant buildings (i.e., 2.5 times containment building height per Reference 13). All releases are assumed to be under the influence of the containment building wake effect, with the exception of the spent fuel building release. The applicable structure relative to building wake effects on the spent fuel building release is the auxiliary building.

For PBNP, release source from the containment building during a large break LOCA will be treated as a diffuse source (i.e., the containment can potentially leak anywhere on the exposed surface). In this accident, the activity released is assumed to be homogeneously distributed throughout the containment building and released at a constant rate from the building surface. In order to have ARCON96 treat the containment surface area as a virtual point source, initial horizontal and vertical diffusion coefficients are approximated. These diffusion coefficients were calculated by dividing the containment width and height by six as directed by RG 1.194 (Reference 13). For PBNP, the containment width is 112.0 ft and height is 130.75 ft. Therefore, the horizontal and vertical coefficients are 5.7 m and 6.6 m, respectively. In addition, the atmospheric dispersion factors were calculated using the shortest distance between the containment building and the control room intake. Both the Unit 1 and Unit 2 χ/Q were calculated in this manner, however, χ/Q values associated with Unit 2 yield a more conservative atmospheric dispersion factor. All other non-containment release paths are treated as point sources.

The specific release point/paths for which χ/Q values are calculated are listed below. These locations generated the most conservative atmospheric dispersion factors for the intended application. For each release location, the receptor is the control room fresh air intake. This receptor location is also used conservatively for unfiltered inleakage. Atmospheric dispersion factors calculated at the north and south control room doors were lower than at the control room air intake structure. Release locations 1 and 2 were used to calculate the concentrations at the north and south doors of the control room at El. 46' for purposes of determining the external cloud shine dose. Figure 1 shows the Unit 1, Unit 2 and common release locations and Figure 2 shows the PBNP site plan.

1. Unit 2 containment wall
2. Auxiliary building vent stack
3. Unit 2 main steam safety valves (A and B)
4. Unit 2 containment façade
5. Unit 2 purge stack

The following assumptions are made for these calculations:

1. The plume centerline from each release is conservatively transported directly over the control room air intake or north and south doors;

2. All releases are assumed to be under the influence of the containment building wake effect, with the exception of the spent fuel building release. The applicable structure relative to building wake effects on the spent fuel building release is the auxiliary building based on the release to receptor orientation;

3. The MSSV/ADV releases (i.e., Unit 2 safeties) are from the approximate center of the discharge vents;

4. The control room air intake χ/Q values are representative of the χ/Q values for the center of the control room, and the north and south doors since the distances and directions from these releases to these receptors are very similar;

5. The ARCON96 default wind direction range of 90°, centered on the direction that transports the gaseous effluents from the release points to the receptors is used in the calculation per Reference 13;

6. The ARCON96 values for surface roughness length (i.e., 0.20 meter) and sector averaging constant (i.e., 4.3) are based on Reference 13;

7. All releases are conservatively treated as ground level as there are no release conditions that merit categorization as an elevated release (i.e., 2.5 times the containment building height, Reference 13) with respect to the PBNP configuration.

The ARCON96 input values for each release point are provided in Tables 8 through 13. The χ/Q values for all release locations are provided in the accident specific input tables.

The meteorological data in the ARCON96 format is provided as Enclosure 6 on CD-ROM. This file was compiled from the PBNP data collected from the primary tower from September of 2000 to September 2005. Each line of the file represents the location identifier (e.g., PTPCH), day of the year (Julian), hour of the day, lower wind speed (mph), lower wind direction, stability class based on temperature difference between the 45 and 10 meter tower levels according to the NRC temperature difference range approach (i.e., °C/100 m), upper wind speed (mph) and upper wind direction. Missing data are represented by the designation of 9's as directed by NUREG/CR-6331.

5.0 Control Room Envelope

5.1 Control Room Licensing Basis

The PBNP control room design was implemented and licensed under site-specific General Design Criterion (GDC) 11, which is similar to the criterion proposed by the AEC in 1967 before the issuance of the GDC in 10 CFR 50, Appendix A. PBNP GDC 11 states that the facility shall be provided with a control room from which actions to maintain safe operational status can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of CR under any credible post-accident condition or as an alternative, access to other areas as necessary to shutdown and maintain safe control of the facility without excessive radiation exposures of personnel. Although this design criterion is applicable in many other areas, such as fire protection, HELB, security, etc., the focus of this section is solely on radiological habitability of the control room.

The control room ventilation system is designed to provide heating, ventilation, air conditioning, and radiological habitability for the control and computer rooms, both of which are within the control room envelope. For radiological habitability, the system is capable of providing control room pressurization to limit inleakage, makeup and recirculation through HEPA and charcoal filters to remove contaminates, and recirculation with or without filtration or makeup. Design and system reviews stemming from the post-TMI initiatives demonstrate that the system is capable of meeting the dose limits of 10 CFR 50 Appendix A GDC-19 as required by NUREG-0737, Item III.D.3.4 while taking credit for use of potassium iodide (KI) to reduce the thyroid dose. The design factors affecting the system's ability to meet the above dose limits include: actuation on a containment isolation or high radiation signal; emergency total filtration flow rate of 4950 cfm ±10%; maintaining a positive pressure during the accident mitigation mode of operation; and meeting minimum filtration efficiencies specified in the test section for the HEPA and charcoal filters.

5.2 Current Control Room Design and Ventilation System (VNCR) Description

The PBNP control room envelope is located in the control building within the turbine building approximately half way between Unit 1 and Unit 2. The control room envelope consists of the control room, the computer room and each room's associated ductwork as it transitions through the mechanical equipment room. The cable spreading room on El. 26' (directly below the control room) and the mechanical equipment room on El. 60' (directly above the control room) are not part of the control room envelope.

Two types of radiation monitors with control functions are located within the control room envelope: an area monitor and a process monitor. The area monitor, (RE-101), located on the

west wall of the control room, is a low-range gamma sensitive G-M tube detector assembly. The process monitor (RE-235) is a scintillation type detector, calibrated to Xe-133, and physically located on the control building roof. The sensing line penetrates the control room supply ductwork downstream of the control room HVAC filter unit. Because noble gases cannot be filtered via HEPA or charcoal, the monitor measurements are relatively unaffected by the filters. A "high" signal from either detector will automatically switch the control room ventilation system (VNCR) from the normal mode of operation to the emergency mode.

The control room ventilation system is currently designed for four modes of operation. Mode 1 is normal operation, mode 2 is 100% recirculation, mode 3 is 25% filtered return air/75% recirculation, mode 4 is 25% filtered outside air/75% recirculation; mode 5 (to be implemented by modification) is 10% filtered outside air/15% filtered return air/75% recirculation. Because modes 2 and 3 are not relied on for the accident analysis presented in this submittal, they will not be discussed further.

For mode 1, one of the two normal supply/recirculation fans (W-13B1 or W-13B2) is started. The fan start opens the outside air damper VNCR-4849C to a predetermined throttled position to supply approximately 1000 cfm of outside air ducted from an intake penthouse located on the roof of the auxiliary building. The outside air and return air from the control room and computer room pass through roughing filter F-43 and cooling coils HX-100A & B before entering one of the normal recirculation fans. Room thermostats and/or humidistats control operation of the chilled water unit supplying the cooling coils. After leaving the normal recirculation fan, the filtered and cooled air passes through separate heating coils, HX-92 and HX-91A & B, and humidifiers, Z-78 and Z-77, to the computer and control rooms respectively. Room thermostats and humidistats also control the operation of the heating coils and humidifiers. Also operating in mode 1 are computer room supplemental air conditioning unit W-107A/HX-190A/HX-191A or W-107B/HX-190B/HX-191B and control room washroom exhaust fan W-15.

When mode 4 is actuated, the return air inlet damper VNCR-4851B to the emergency fans is closed and outside air supply damper VNCR-4851A opens. This allows approximately 4950 cfm of makeup air to pass through filter F-16 and the emergency fan to the suction of the normal recirculation fan, ensuring a positive pressure is maintained in the control and computer rooms to prevent inleakage. After modification, mode 4 will only be initiated manually from panel C-67 in the control room.

In light of recent industry concerns with regard to control room habitability, initiatives have been taken to further increase system reliability, improve program implementation, gain safety margin, and increase the integrity of the control room HVAC system by leak tightening the envelope to reduce the potential areas for unfiltered air infiltration. Improvements to the system were made by the replacement of dampers on the periphery of the control room envelope (CRE) with bubble-tight dampers (extremely low leakage dampers) and hardcasting the seams of portions of the CRE ductwork. The hardcasting is a sealant applied to the seams of the ductwork consisting of a fibrous material bonded with an epoxy-like adhesive material.

Other modifications completed to date include installation of a new balance damper and bubble tight isolation damper upstream of the cable spreading room outside air intake isolation, installation of a new bubble tight damper at the discharge of the control room washroom exhaust fan, installation of three new bubble tight dampers for the control room, computer room, and cable spreading room smoke and heat exhaust fan isolation, upgrades to the control room backup instrument air system, replacement of existing control room washroom exhaust

fan with a direct drive fan, and improved differential pressure indication between the control room and the turbine building.

5.3 Proposed Design Changes to the Control Room Envelope (CRE)

The proposed AST accident analyses presented in Section 6 do not assume the current mode 4 configuration for accident mitigation. Instead, the VNCR is assumed to operate with filtered return air in addition to filtered makeup air. This configuration is referred to as mode 5 (emergency HEPA/charcoal filtered outside air and HEPA/charcoal filtered return air mode). Operation in this mode is similar to mode 4 except that the return air inlet damper VNCR-4851B to the emergency fans opens. This allows a combination of outside air (≤2500 cfm) and return air to pass through the emergency HEPA/charcoal filter unit to the suction of the control room recirculation fan for a total flow rate of 4950 cfm ±10%. Mode 5 will be automatically initiated (after modification) by a containment isolation signal, or by a high radiation signal from the control room area monitor RE-101, or by a high radiation signal from the process monitor RE-235 located in the supply duct to the control room. This mode of operation can also be manually initiated from the control room. Operation in mode 5 is the assumed mode of operation for the control room habitability analyses, and is therefore, the only mode of operation addressed by TS Limiting Condition for Operation (LCO).

The total filter flow rate of 4950 cfm ±10% remains the acceptance criteria for the filter test surveillance (TS 5.5.10). Sensitivities performed on this tolerance band demonstrated that the minimum total filter flow (4455 cfm) was limiting with respect to operator dose. Therefore, the analyses assume the emergency mode of operation for the VNCR is 2500 cfm filtered makeup air with 1955 cfm filtered return air.

Additional shielding will be incorporated outside the control room envelope. Modifications will add concrete shielding to the control room F-16 filter, entrances and windows. The LOCA analysis, which is the bounding accident analysis, takes direct credit for these modifications in the attenuation of radiation resulting from direct containment shine, external cloud due to leakage (both containment and ECCS) and filter shine.

This LAR is contingent on implementation of the plant modifications discussed above: 1) new control room emergency filtration system mode 5 accident mode operation, including operating procedure changes, and 2) shielding modifications to the control room envelope.

6.0 Radiological Accident Analysis

As discussed in Section 1.0, a full implementation of the AST, as defined in Section 1.2.1 of Reference 2, is proposed for PBNP Units 1 and 2. The methodology and scenarios used in the existing design basis accident analyses discussed in the PBNP FSAR are being updated to reflect the guidance provided in Regulatory Guide 1.183 (Reference 2). These analyses also include revised control room atmospheric dispersion factors developed using ARCON96 (Reference 4). The offsite (EAB and LPZ) and control room dose analyses for the following design basis accident have been reanalyzed using the AST as allowed by 10 CFR 50.67:

1. Loss of Coolant Accident (LOCA)
2. Steam Generator Tube Rupture (SGTR)
3. Loss of Coolant Flow (Locked Rotor - LR)
4. Rupture of a Steam Pipe (Main Steam Line Break - MSLB)
5. Rupture of a Control Rod Mechanism Housing – RCCA Ejection (CRDE)
6. Fuel Handling Accident (FHA)
7. Reactor Vessel Head Drop Accident (RVHD)

The FHA is already approved to use the AST methodology. (Reference 19) It is represented in this amendment for consistency, as well as, to present a change in the dose to the control room due to the proposed control room HVAC emergency mode change and a change in the atmospheric dispersion factors for the control room.

In addition, higher gap fractions were applied for the fraction of the fuel that does not meet the limits on burnup and linear heat generation rate, following the method approved by the NRC for Kewaunee Power Station (ML070430020). The gap fractions are those from Safety Guide 25 with the value for I-131 adjusted consistent with the recommendation in NUREG/CR-5009.

The worst 2 hour period dose at the EAB and the dose at the LPZ for the duration of the release are calculated for each of these events based on postulated airborne radioactivity releases. This represents the post-accident dose to the public due to inhalation and submersion for each of these events. Due to distance from the plant and plant shielding, the dose contribution at the offsite locations due to direct shine from contained sources is considered negligible for all of the accidents.

The 0 to 30 day dose to an operator in the control room due to airborne radioactivity releases is developed for all of the referenced design basis accidents. This represents the post-accident dose to the operator due to inhalation and submersion. The control room shielding design is based on the LOCA, which represents the worst case DBA relative to radioactivity releases. The direct shine dose due to contained sources and the external cloud is included in the control room doses reported for the LOCA. The contribution of shine due to other accidents is not explicitly determined for non-LOCA accidents. For these accidents, it is assumed that the shine dose contribution is negligible.

The doses provided in the evaluations are generally reported to the nearest 0.1 rem.

6.1 Large Break Loss of Coolant Accident Doses (LOCA)

Radiological consequences due to a LOCA are due to a postulated abrupt failure of the main reactor coolant pipe in which the emergency core cooling features fail to prevent the core from experiencing significant degradation (i.e., melting). Activity from the core is released to the containment and from there released to the environment by containment leakage and leakage from the ECCS.

The reanalysis of the LOCA offsite and control room doses for PBNP uses the following RG 1.183 source term characteristics in place of those identified in TID-14844 (Reference 1) and RG 1.4 (Reference 11):

- Iodine chemical species
- Fission product release timing
- Fission product release fractions
- Fission product groups

Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 18. Activity is released from the fuel into the containment using the timing and release fractions from Table 15 and Table 16. The analysis considers the release of activity from the containment via containment leakage. In addition, once the recirculation mode of the ECCS is established, activity in the sump solution may be released to the environment by leakage from ECCS equipment into the auxiliary building. No credit for auxiliary building vent stack filtration is taken.

The offsite and control room doses are the sum of the doses resulting from each of the postulated release paths. The EAB dose is reported for the worst 2 hour period. This is determined by calculating the dose during various time intervals. The doses to the LPZ and control room are reported for the duration of the accident (i.e., 30 days). The following sections address topics of significant interest.

Source Term

The reactor coolant activity is assumed to be released over the first 30 seconds of the accident. The release of the reactor coolant activity is not explicitly modeled, because the activity in the coolant is insignificant when compared to the release from the core and is not included in the analysis.

The use of RG 1.183 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 core occurs over a 1.8 hour interval. The gap release phase occurs in the first half hour and the release from the melted fuel occurs over the next 1.3 hours.

Instead of considering only the release of iodines and noble gases, a wide spectrum of nuclides is taken into consideration. Table 17 lists the nuclides being considered for the LOCA with core melt (eight groups of nuclides). Tables 15 and 16 provide the fission product release fractions and the timing/duration of releases to the containment as assumed in RG 1.183.

Instead of the iodine being primarily in the elemental form, the iodine is mainly in the form of cesium iodide (CsI), which exists as particulate. The iodine characterization from RG 1.183 is provided in Table 14.

For the containment leakage analysis, all activity released from the fuel is assumed to be in the containment atmosphere until removed by sprays, sedimentation, radioactive decay or leakage from the containment. For the ECCS leakage analysis, 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.

Containment Modeling

The containment building is modeled as two discrete volumes: sprayed and unsprayed. The volumes are conservatively assumed to be mixed only by the containment fan coolers. The containment volume is 1.0E6 ft^3 with a sprayed fraction of 58.2 percent of the total (5.82E5 ft^3) (Table 18).

The containment is assumed to leak at the proposed TS leak rate of 0.2 percent per day (Bases for TS 3.6.1) for the first 24 hours of the accident and then to leak at half that rate (0.1 percent per day) for the remainder of the 30 day period following the accident.

Removal of Activity from the Containment Atmosphere

The reduction of activity available for release to the environment depends on the chemical form. The removal of elemental iodine from the containment atmosphere is accomplished only by containment sprays and radioactive decay. The removal of particulates from the containment atmosphere is accomplished by containment sprays, sedimentation, and radioactive decay. The noble gases and the organic iodide are subject to removal only by radioactive decay.

One train of the containment spray system is assumed to operate in the injection mode following the LOCA. When the RWST drains to a predetermined level, the operators switch to recirculation of the sump liquid to provide a source to the sprays. The minimum injection spray duration until the level is reached is 60 minutes. The switchover is assumed to take 20 minutes. During these 20 minutes, the analysis does not credit any spray removal in the containment. Sensitivity analyses determined that (for the RG 1.183 release model) the minimum time to switchover to recirculation is conservative. The analysis conservatively assumed the minimum injection spray time of 60 minutes. The analysis assumed that the recirculation spray valves operate for 4 hour duration.

Current FSAR radiological accident analyses do not take credit for operation of the containment spray system in the containment sump recirculation phase. The dose projections prepared in support of this submittal assume that containment spray is maintained throughout the injection phase of a LOCA, and continued during the early portions of the recirculation phase with no more than a 20 minute interruption. The ability to maintain spray during the early recirculation phase is essential, as this is the period of highest iodine evolution from a damaged core.

To support this assumption, it will be necessary for PBNP to alter the existing emergency operating procedure(s) to direct continued containment spray while on sump recirculation if containment radiological conditions and/or core damage indicates it is required. Validation of the procedures will ensure that no more than a 20 minute interruption in containment spray flow is incurred.

The physical capability of the supporting systems will also need to be modified to ensure this capability is maintained. A loss of a single train of RHR would require supplying both core deluge and containment spray from the same RHR pump. If a concurrent loss of the (non-safety related) instrument air system occurred, the RHR heat exchanger outlet valve would fail open. In this configuration, the operating RHR pump would not have sufficient net positive suction head (NPSH) for the high flow rates projected. To alleviate this possibility, the RHR, containment spray system and support systems will be modified to ensure that no combination of single failures and failures of non-safety related equipment can challenge RHR pump operations. Also, this will likely include modifying the containment spray pump discharge motor-operated valves to permit throttling. The supporting analyses have derated the design spray flow rates to reflect a reduction in spray flow during the recirculation phase consistent with RHR pump NPSH limitations.

The containment spray system piping, pumps, seals, etc., will be fully capable of supporting continued spray during recirculation once the necessary modifications to limit total RHR flow and the associated emergency operating procedure changes have both been completed. This license amendment request is contingent on implementation of the containment spray and RHR plant modifications and associated emergency operating procedure changes.

Containment Spray Removal of Elemental Iodine

The Standard Review Plan (Reference 12) 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:

- 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 of the removal rate was specified as 20 hr^{-1} for this model.

For PBNP, a two-loop design, an elemental iodine removal coefficient of 20 hr^{-1} was determined based on the model suggested in NUREG-0800 (SRP 6.5.2). When spray valves are operating in the recirculation phase the elemental removal coefficient is reduced to 10 hr^{-1} to address the loading of the recirculating solution with elemental iodine.

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 decontamination factor or DF of 200). With the RG 1.183 source term methodology this is considered as being 0.5 percent 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 at 2.66 hours.

Containment Spray Removal of Particulates

Particulate spray removal is determined using the model described in Reference 12. The first order spray removal rate constant for particulates may be written as follows:

$\lambda_p = 3hFE / 2Vd$

Where:

- h = Drop Fall Height, ft
- F = Spray Flow Rate, ft^3/hr
- V = Volume Sprayed, ft^3
- E = Single Drop Collection Efficiency
- d = Drop Diameter, ft

The E/d term depends upon the particle size distribution and spray drop size. From Reference 12, it is conservative to use 10 m^{-1} (3.05 ft^{-1}) for E/d until the point is reached when the inventory in the atmosphere is reduced to 2 percent of its original amount (DF of 50) at which time it is reduced to 1.0 m^{-1}. With the RG 1.183 source term methodology this is considered as being 2 percent of the total inventory of 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 18. When the airborne inventory drops to 2 percent of the total particulate iodine released to the containment (DF of 50), this removal coefficient is reduced by a factor of 10. In the analysis this occurs at 3.29 hours.

Sedimentation Removal of Particulates

During spray operation, credit is taken for sedimentation removal of particulates in the unsprayed region. After sprays are terminated (and during the 20 minute switchover from injection to recirculation when sprays are not credited), credit for sedimentation is taken in both the sprayed and unsprayed regions.

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 15). The CSE tests determined that there was a significant sedimentation removal rate even with a relatively low aerosol concentration. From Reference 15, even at an air concentration of 10 $\mu g/m^3$, the sedimentation removal coefficient was above 0.3 hr^{-1}. With 2.0 percent of particulates remaining airborne at the end of the credited spray removal period, there would be more than 10,000 $\mu g/m^3$ and an even higher sedimentation rate would be expected. As noted above, the DF of 50 occurs at 3.29 hr. For the analysis, the sedimentation removal coefficient is conservatively assumed to be only 0.1 hr^{-1}. This value for sedimentation removal of particulates has been accepted by the NRC for Indian Point Unit 2 and Shearon Harris in the Safety Evaluation Reports for the application of the Alternative Source Term methodology. It is also conservatively assumed that sedimentation removal does not continue beyond a DF of 1000. A DF of 1000 is reached at 29.12 hours.

ECCS Leakage

When ECCS recirculation is established following the LOCA, leakage is assumed to occur from ECCS equipment outside containment. This leakage goes into the auxiliary building and is released to the environment, unfiltered, through the auxiliary building vent stack. Recirculation is conservatively initiated at 0 minutes. The assumption of the ECCS leakage beginning at 0 minutes is not consistent with the assumption of injection spray termination in the containment leakage portion of the analysis. However, beginning the ECCS leakage at 0 minutes adds conservatism to the dose consequences. The leakage continues for the 30 day period following the accident considered in the analysis.

Activity enters the sump and flows out of containment in the ECCS recirculation flow and is released to the environment through leakage from the ECCS. Only iodine is released through this pathway since the noble gases are not assumed to dissolve in the sump and particulates would remain in the water of the ECCS leakage. It is assumed that the iodine is instantaneously and homogeneously mixed in the primary containment sump water at the time of release from the core.

The total ECCS recirculation leakage into the primary auxiliary building modeled in the analysis is 0.2113 gpm (i.e., the assumed value of 400 cc/min total ECCS leakage outside the containment is doubled consistent with RG 1.183 guidance).

Control Room Direct Shine Dose Due to External Cloud and Contained Sources

The dose contribution in the control room due to direct shine from the external cloud and from contained sources is addressed. The external cloud contribution includes containment leakage and ECCS leakage. The contained sources include shine from the containment structure and the control room HVAC filter. The 30 day deep dose equivalent (DDE) to a control room operator due to the airborne source in containment, the passing plume source and the control room filter source is calculated. The integrated dose is calculated as a function of distance from the south door and north door openings and from the filter source.

The analysis takes credit for shielding modifications at the south door, east window, and work control center (WCC) outer walls located on the north end of the control room. In addition, it is assumed that shielding is provided on the floor of the mechanical equipment room near the control room HVAC filter unit (located above the control room proper).

The containment shine source includes all of the airborne source above the operating floor of El. 66'. This volume includes the volume in the cylindrical section and the volume in the spherical-toroidal dome. The total volume is modeled as an equivalent cylindrical source. Only the major intervening shielding is credited (e.g., the containment liner/wall, control room walls and ceiling, etc.). For the south control room door location, a Unit 1 LOCA is assumed and the additional shielding is in place. For the north control room door location, a Unit 2 LOCA is assumed, as well as, additional shielding of the work control center (WCC) windows.

The external plume is assumed to be uniformly distributed source from El. 26' to approximately 1000 meters above the ground. The plume source is divided into three parts to assess the dose due to the south and north door locations of the control room as well as proximity to the east window. A Unit 1 LOCA is assumed for the south door and east window, and a Unit 2 LOCA is assumed for the north door location. Similar to the direct shine contribution, only the major intervening shielding is credited (e.g., control room walls and ceiling, turbine building floors etc.).

The control room HVAC filter shine dose is calculated on the elemental, organic and particulate iodine releases from containment. Only the elemental and organic forms of iodine are considered to be part of the ECCS leakage source term. The ECCS release path assumes that the particulate form of iodine remains in the water. Although the present control room emergency mode is filtered outside air with unfiltered recirculation, the filter shine dose is based on the control room HVAC operating in a mode that allows for filtered recirculation. This maximizes the loading of the filter for the duration of the accident.

All doses are conservatively calculated at 10 ft. from the control room doors and 10 ft. horizontal distance from the control room HVAC filter. The external cloud and contained source doses are listed in the Results and Conclusions section below.

Control Room Isolation

In the event of a large break LOCA, the safety injection (SI) setpoint will be reached shortly after event initiation. The SI/containment isolation signal causes the control room heating, ventilation and air conditioning (HVAC) to switch from the normal operation mode to the post-accident mode of operation. It is assumed that the SI setpoint is reached immediately at the start of the event and a conservative 60 second delay time for switching from normal to post-accident operating mode (filtered recirculation with filtered fresh air intake) is modeled.

Analysis Sensitivities on Unfiltered Inleakage

An analysis was conducted to determine the maximum amount of unfiltered inleakage the design basis LOCA accident could tolerate without exceeding 5 rem TEDE (including an allowance for shine doses) and without crediting the ingestion of KI. This study assumed a reactor power of 1540 MWt plus the 10 CFR 50 Appendix K uncertainty of 0.6%, and the VNCR mode 5 configuration (2500 cfm unfiltered make up air with 1955 cfm filtered return air). The offsite doses are 12.4 rem at the exclusion area boundary (EAB) in the worst 2 hour interval (0.6-2.6 hours) and 1.7 rem at the low population zone (LPZ). The control room doses are 4.6 rem (excluding shine dose) with an unfiltered inleakage of 125 cfm.

Acceptance Criteria

The exclusion area boundary (EAB) and low population zone (LPZ) dose acceptance criteria for a LOCA is 25 rem TEDE per RG 1.183. This is the 10 CFR 50.67 limit. The acceptance criterion for the control room dose is 5.0 rem TEDE per 10 CFR 50.67. The exclusion area boundary doses are calculated for the worst 2 hours. The low population zone and control room doses are calculated for 30 days.

Results and Conclusions

The large break LOCA doses are:

Exclusion Area Boundary	13.5 rem TEDE
Low Population Zone	1.9 rem TEDE
Control Room – All Pathways	4.8 rem TEDE

The acceptance criteria are met.

The exclusion area boundary dose reported is for the worst 2 hour period, determined to be from 0.6 hours to 2.6 hours.

6.2 Steam Generator Tube Rupture Accident Doses (SGTR)

The evaluation of the radiological consequences of a SGTR assumes that the reactor has been operating at the TS limits for primary coolant activity (TS 3.4.16) and primary-to-secondary leakage (TS 3.4.13) for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant.

A double-ended rupture of a single steam generator tube is assumed to occur. At the start of the accident, radionuclides from the primary coolant enter the steam generator, via the ruptured tube and primary-to-secondary leakage, and are released to the atmosphere through the condenser air ejector exhaust via the auxiliary building vent stack (ABVS) prior to reactor trip. The primary-to-secondary break flow results in depressurization of the reactor coolant system (RCS). It is assumed that reactor trip and safety injection (SI) are automatically initiated simultaneously on low pressurizer pressure. A loss of offsite power is assumed concurrent with the reactor trip; therefore, use of the condenser is lost and the steam is released via the steam generator safety or atmospheric dump valves (ADVs).

Following reactor trip and SI actuation, it is assumed that the RCS pressure stabilizes at the equilibrium point where the incoming SI flow rate equals the outgoing break flow rate. 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 that the operators have completed the actions necessary to terminate the steam release from the ruptured steam generator. Pressure between the ruptured steam generator and the primary system is such that the ruptured steam generator is not overfilled.

It was assumed that the break flow is terminated by operator action at 30 minutes to isolate the ruptured SG. This does not constitute a requirement that the operators demonstrate the ability to terminate break flow within 30 minutes from the start of the event and it is recognized that the

operators may not be able to terminate break flow within 30 minutes for all postulated SGTR events. The purpose of the calculation is to provide conservatively high mass-transfer rates for use in the radiological consequences analysis. This was achieved by assuming a constant break flow at the equilibrium flow rate, with a constant flashing fraction that does not credit the plant cooldown, for a relatively long time period. Thirty minutes was selected for this purpose. This modeling is consistent with the SGTR analysis currently presented in Section 14.2.4 of the Final Safety Analysis Report (FSAR).

Westinghouse has performed a SGTR analysis with the operator response time increased from 30 minutes to 46 minutes. The analysis also took into account the effect of charging flow and considered the potential for SG overfill. The analysis demonstrates that the results for the 46 minute release period with a more precisely modeled mass-release analysis are bounded by the 30 minute duration constant break flow model. Thus, the analysis justifies extending the allowable time from 30 to 46 minutes for operator response in the affected emergency operating procedures while the dose analysis remains conservatively based on the 30 minute release model.

After 30 minutes, it is 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 residual heat removal (RHR) system operating conditions. During post-SGTR cooldown, the pressure in the ruptured steam generator is assumed to be decreased by the backfill method in which core decay heat and RCS fluid energy is dissipated by releasing steam from the intact steam generator. This is the preferred approach since it minimizes the radioactivity released to the atmosphere. It is assumed that the plant cooldown to RHR operating conditions is accomplished within 8 hours after initiation of the SGTR and that steam releases are terminated at this time. 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 and from 2 to 8 hours.

A portion of the break flow will flash directly to steam upon entering the secondary side of the ruptured steam generator. This has not been previously considered in the PBNP SGTR analyses. The analysis performed for this submittal incorporates a break flow flashing fraction. Since a transient break flow calculation is not performed, 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 quantity of radioactivity released to the environment due to a SGTR depends upon primary and secondary coolant activity, iodine spiking effects, primary-to-secondary break flow, break flow flashing, attenuation of iodine carried by the flashed portion of the break flow, partitioning of iodine between the liquid and steam phases, the mass of fluid released from the generators and liquid-vapor partitioning in the turbine condenser hot well. All of these parameters were conservatively evaluated for a design basis double ended rupture of a single tube.

The releases conservatively bound the assumed power level of 1683 MWt including power measurement uncertainties. The resulting offsite and control room doses are calculated in this section. This section includes the methods and assumptions used to analyze the radiological consequences of the SGTR event as well as the calculated results.

Input Parameters and Assumptions

Input parameters and assumptions are provided in Table 19.

Source Term Assumptions

The concentration of noble gases and iodines in the RCS has been calculated based on a one percent fuel defect level for the assumed power level of 1683 MWt. The concentration data is presented in Table 6 and is used in the SGTR analysis. The initial concentration of noble gases in the reactor coolant corresponds to the TS (TS 3.4.16) limit of 100/E-bar. The conversion from the one percent fuel defect values in Table 6 to DE I-131 employs dose conversion factors (DCF's). These DCF's are used in the calculation of the initial RCS iodine concentrations. The thyroid dose conversion factors used in the analysis are from Reference 6 and are presented in Table 20.

The iodine concentrations in the primary and secondary system, prior to and following the SGTR, are based upon pre-accident and accident-initiated iodine spikes as outlined in RG 1.183, Appendix F (Reference 2).

Pre-accident Spike - A reactor transient has occurred prior to the SGTR and has raised the primary coolant iodine concentration to a conservative value of 60 μCi/gm of DE I-131. The TS 3.4.16 limit for a transient is 50 μCi/gm of DE I-131. Table 21 provides iodine coolant activity corresponding to 60 μCi/gm of DE I-131.

Accident-Initiated Spike - The primary coolant iodine concentration is initially at the proposed TS 3.4.16 limit, specified as 0.5 μCi/gm of DE I-131. Following the primary system depressurization and reactor trip associated with the SGTR, an iodine spike is initiated in the primary system. This spike increases the iodine appearance rate from the fuel to the coolant to a value 335 times greater than the release rate corresponding to the initial primary system iodine concentration consistent with RG 1.183, Appendix F. The spike appearance rates are provided in Table 22. This release rate (the equilibrium iodine appearance rate) is calculated to match the rate of iodine removal from the RCS. Iodine removal from the RCS is the combination of decay, leakage, and cleanup.

The Nuclear Safety Advisory Letter (NSAL) (Reference 14) identified non-conservative assumptions that had been used in the calculation of accident-initiated iodine spiking rates in the primary coolant. The issues identified in this NSAL were applicable to the PBNP analysis. To address the issues identified in the NSAL, the conservative spike model calculates the equilibrium iodine appearance rates based on a letdown flow of 90 gpm with perfect cleanup. This flow is conservatively increased by 10 percent to cover uncertainties in the flow. In addition, a total of 11 gpm leakage from the RCS allowed by the TS 3.4.13 limit (which also removes iodine from the RCS) is considered in the calculations. The effective letdown flow is 110 gpm with the spike model suggested by the NSAL. The 110 gpm is the total of 90 gpm letdown flow with perfect cleanup increased by 10 percent to 99 gpm (to cover uncertainty) and 11 gpm allowable leakage from the RCS.

The initial RCS iodine activities used in the analysis are presented in Table 21. These values serve as the basis for the iodine appearance rate calculations. The iodine appearance rates used in the analysis are presented in Table 22. For the accident initiated spike scenario, the iodine spike (i.e., the appearance rate times 335, as discussed above) persists for 8 hours from

the start of the event per RG 1.183. After this point there is no activity available for release from the gap.

The initial secondary coolant iodine concentration is 0.1 μCi/gm of DE I-131 (proposed TS 3.7.13). The chemical form of iodine released from the steam generators is assumed to be 97 percent elemental and 3 percent organic. Noble gases are not present in the secondary system at the start of the event.

Dose Calculation Assumptions

Offsite power is assumed to be lost at reactor trip. This assumption was used in the thermal hydraulic analysis to maximize break flow and steam release from the ruptured steam generator. Prior to reactor trip, activity released through the condenser air ejector exhaust is included for completeness and to establish the limiting case. An iodine partition factor of 0.01 is assumed for this release path. Although the air ejector exhausts through the auxiliary building vent stack to the environment, the atmospheric dispersion factors associated with the Unit 2 safety valves is used to determine the concentration of this release path at the control room intake. After reactor trip and loss of offsite power, flow to the condenser is isolated. This condenser iodine partition factor is consistent with the RG 1.183 (Reference 2) steam/water partition coefficient for SGs.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used.

The iodine transport model used in this analysis accounts for break flow flashing, steaming and partitioning. The model assumes that a fraction of the iodine carried by the break flow becomes airborne immediately due to flashing and atomization. Droplet removal by the dryers is conservatively neglected. The fraction of primary coolant iodine that is not assumed to become airborne immediately mixes with the secondary water, and is assumed to become airborne at a rate proportional to the steaming rate. The steam/water partition coefficients discussed above are used.

In the iodine transport model, the time dependent iodine removal efficiency for scrubbing of steam bubbles as they rise from the rupture site to the water surface was not calculated and was conservatively neglected.

All of the iodine in the flashed break flow is assumed to be transferred instantly out of the steam generator to the atmosphere.

The issue of tube bundle uncovery was considered in a Westinghouse Owners Group (WOG) program (Reference 16). The WOG program concluded that the effect of tube uncovery is essentially negligible for the limiting SGTR transient. The WOG program concluded that the steam generator tube uncovery issue could be closed without any further investigation or generic restrictions. The NRC review of the WOG submittal (Reference 17) concluded "... the Westinghouse analyses demonstrate that the effects of partial steam generator tube uncovery on the iodine release for SGTR and non-SGTR events is negligible. Therefore, we agree with your position on this matter and consider this issue resolved."

Since there is no penalty taken for tube uncovery and no iodine scrubbing is credited, the location of the tube rupture is not significant for the radiological analysis. The thermal and

hydraulic analysis has conservatively addressed the issue of the location of the tube rupture in the calculations of break flow flashing.

All noble gases in the break flow and primary-to-secondary leakage are assumed to be transferred instantly out of the steam generator to the atmosphere. Iodine and noble gas decay constants are presented in Table 7. These decay constants were calculated from half-lives given in Reference 18.

Offsite atmospheric dispersion factors (χ/Qs) for accident analysis and breathing rates are provided in Table 3. The offsite and control room breathing rates and control room occupancy factors are consistent with RG 1.183.

Mass Transfer Assumptions

A total primary-to-secondary leak rate is assumed to be 0.7 gpm. The leak is assumed to be distributed with 0.35 gpm to the intact steam generator and 0.35 gpm to the ruptured steam generator. The leakage to the intact steam generator is assumed to persist for the duration of the accident. The density for this leakage is 47 lb_m/ft^3, consistent with the PBNP procedure for determining primary-to-secondary leakage.

A sensitivity study was performed for the thermal-hydraulic analysis in order to determine the limiting break flow and steam releases for the first 30 minutes of the accident. This sensitivity study considered both types of SGs (Westinghouse Models 44F and Δ47F) varied the amount of tube plugging (0 to10 percent), RCS average temperature (558.1°F to 574°F), and the lower pressurizer pressure SI actuation setpoint (1880 or 1730 psia). Tube plugging level is assumed to be 0 percent steam generator tube plugging (SGTP) to reflect higher steam pressure and temperatures related to the clean steam generator conditions and 10 percent SGTP to reflect lower steam pressure and temperature at the maximum tube plugging condition. A single calculation was performed to calculate the long-term steam releases from the intact steam generator for the time intervals of 0 to 2 hours and 2 to 24 hours.

The integrated tube rupture break flow, break flow flashing fractions, integrated atmospheric steam releases and feedwater flows for the offsite radiological analysis are summarized in Table 19. The maximum steam release calculation for the initial 30 minutes is based on the results of the case modeling the Model Δ47F steam generator, 0 percent tube plugging (higher initial secondary temperature), 574°F RCS average temperature, and a low pressurizer pressure SI setpoint of 1880 psia.

The largest contribution to the offsite doses comes from the release of flashed break flow. Flashed break flow after reactor trip and the assumed loss of offsite power (and condenser) contributes more to the calculated doses than that released prior to trip. Therefore, the break flow results are selected to maximize post-trip break flow. The maximum primary-to-secondary break flow is independent of the steam generator design and is based on the analysis for 10 percent tube plugging (lower initial secondary pressure), 558.1°F RCS average temperature, and an elevated pressurizer pressure SI setpoint of 1880 psia (earlier reactor trip). For an SGTR event, the most significant amount of radioactivity released to the atmosphere is dependent on the amount of flashed break flow and the amount of steam released through the steam system safety valves of the ruptured steam generator. Likewise, a greater break flow results in higher contamination of the secondary side, which results in a larger concentration of activity released via the safety valves. Consequently, the worst radiological consequences result from the SGTR case with the greatest amount of flashed break flow and steam released.

Maximum primary-to-secondary break flow, flashed break flow, and steam releases represent bounding values which are conservative for the offsite dose evaluation.

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, although a lower pressure would have a higher heat of vaporization that would result in less flashing. The highest possible pre-trip flashing fraction based on the range of operating conditions covered by this analysis is for a case with a hot leg temperature of 605.5°F, initial RCS pressure of 2250 psia and initial secondary pressure of 640 psia. (This represents a conservative combination of values from different operating condition cases.) All cases consider the same post-trip RCS pressure of 1536.7 psia and post-trip SG pressure of 926.1 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 temperature of 605.5°F. It is conservatively assumed that the hot leg temperature is not reduced for the 30 minutes in which break flow is calculated.

The auxiliary feedwater flow resulting from this analysis is 37.55 lbm/sec. This value represents a minimum flow rate from all the operating pumps assuming no single failure and is calculated based on the mass of the feedwater and the reactor trip time. The SGTR event is typically not considered a limiting transient with respect to the auxiliary feedwater system since the analysis does not consider any failure in the auxiliary feedwater system.

Control Room Isolation

The control room HVAC begins in normal mode. Actuation of the emergency mode is conservatively assumed to occur when the SI/containment isolation actuation setpoint is reached at 147.74 seconds rounded up to 150 seconds (approximately 2.46 minutes). Based on the release during the first moments of the accident, the source term is large enough that the radiation monitor alarm setpoint would have been reached within the first second post accident. In addition, a delay of 60 seconds is assumed to account for HVAC configuration alignment, e.g., damper position changes.

Acceptance Criteria

The doses at the exclusion area boundary (EAB) and the low population zone (LPZ) for a SGTR with an assumed pre-accident iodine spike must be within the RG 1.183 acceptance criterion of 25 rem TEDE. The doses at the EAB and the LPZ for a SGTR with an assumed accident-initiated iodine spike must be within the RG 1.183 acceptance criterion of 2.5 rem TEDE. The doses in the control room must be less than the 10 CFR 50.67 dose acceptance criterion of 5 rem TEDE.

The EAB doses are calculated for the worst 2 hours. The LPZ doses are calculated up to the time all releases are terminated, which is the RHR cut-in time (8 hours) used in the thermal and hydraulic analysis. The control room doses are calculated for 30 days.

Results and Conclusions

The pre-accident iodine spike doses are:

Exclusion Area Boundary	2.0 rem TEDE
Low Population Zone	0.2 rem TEDE
Control Room	1.9 rem TEDE

The accident initiated iodine spike doses are:

Exclusion Area Boundary	0.6 rem TEDE
Low Population Zone	0.1 rem TEDE
Control Room	0.5 rem TEDE

The acceptance criteria are met.

The exclusion area boundary dose reported is for the worst two hour period, determined to be from 0 to 2 hours.

6.3 Locked Rotor Accident Doses (LR)

An instantaneous seizure of a reactor coolant pump rotor is assumed to occur. This rapidly reduces flow through the affected reactor coolant loop. Fuel clad damage may be predicted to occur due to departure from nucleate boiling 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 dump valves (ADV) or main steam safety valves (MSSV). In addition, iodine activity is contained in the secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

Input Parameters and Assumptions

The analysis of the locked rotor radiological consequences uses the analytical methods and assumptions outlined in the RG 1.183, Appendix G. Input parameters and assumptions are provided in Table 23.

It is assumed that 100 percent of the fuel rods in the core suffer damage as a result of the locked rotor sufficient that all of their gap activity is released to the reactor coolant system. Eight percent of the total I-131 core activity is in the fuel-cladding gap. Ten percent of the total Kr-85 core activity is in the fuel-cladding gap. Five percent of other iodine isotopes and other noble gases and 12 percent of the total core activity for alkali metals are assumed to be in the fuel-cladding gap. The fuel clad gap activity fractions are discussed in Section 3.3 of this report.

The iodine activity concentration of the primary coolant at the time the locked rotor occurs is assumed to be equivalent to the proposed TS 3.4.16 limit of 0.5 μCi/gm of DE I-131 and is given in Table 21. The initial concentration of noble gases and alkali metals in the reactor coolant is given in Table 6 and is based on 1 percent defective fuel, which corresponds to the TS (TS 3.4.16) limit of 100/E-bar.

The iodine activity concentration of the secondary coolant at the time the locked rotor occurs is assumed to be equivalent to the proposed TS 3.7.13 limit of 0.1 μCi/gm of DE I-131. The alkali metal activity concentration of the secondary coolant at the time the locked rotor occurs is assumed to be 20 percent of the primary side alkali metal concentration.

As discussed in Section 2.2, the amount of primary-to-secondary SG tube leakage is assumed to be 0.7 gpm total (i.e. 500 gpd per SG). The density for this leakage is 47 lb_m/ft^3.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. Per RG 1.183, the retention of particulates in the SG is limited by moisture carry over; therefore, an alkali metal partition factor (Ci alkali metals/gm steam)/(Ci alkali metals/gm water) in the intact SG of 0.0025 is used. This value is representative of the estimated moisture carryover of the Unit 1 SGs. The steam quality of the Unit 1 SGs is 99.75%, based on a full power, current power level of 1540 MWt and is bounding for both units.

Because the accident analysis is performed for post-trip low power conditions, this value remains conservative for the assumed power level (i.e., 1683 MWt). These partition factors are applied to the primary-to-secondary leakage present in the intact SG. All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

For PBNP, it was assumed that plant cooldown to RHR operating conditions can be accomplished within 8 hours after initiation of the locked rotor event. At 8 hours after the accident, the RHR system is assumed to be placed into service for heat removal and there is no further steam release to the atmosphere from the secondary system. A primary and secondary side mass and energy balance was used to calculate the steam released from the steam generators from 0 to 2 hours, and from 2 to 8 hours. The releases conservatively bound the assumed power of 1683 MWt including uncertainties.

The limiting two hour dose interval at the EAB was determined by performing calculations for various two hour intervals out to 8 hours. The highest calculated dose is reported. The LPZ doses are calculated up to the time all releases are terminated, which is the residual heat removal cut in time assumed to be 8 hours. The control room doses are calculated for 30 days.

Control Room Isolation

The control room HVAC is switched to the post-accident mode after receiving a high radiation ventilation system line monitor signal. This signal is reached almost immediately, however a conservative time of 2 minutes was assumed to switch the control room to the post-accident mode.

Acceptance Criteria

The EAB and LPZ dose acceptance criterion for a locked rotor is 2.5 rem TEDE per RG 1.183. This is 10 percent of the 10 CFR 50.67 limit. The control room dose acceptance criterion is 5.0 rem TEDE per 10 CFR 50.67.

The EAB doses are calculated for the worst 2 hours. The LPZ doses are calculated up to the time all releases are terminated, which is the residual heat removal cut-in time (8 hours). The control room doses are calculated for 30 days.

Results and Conclusions

The locked rotor doses are:

Exclusion Area Boundary	2.2 rem TEDE
Low Population Zone	0.41 rem TEDE
Control Room	3.6 rem TEDE

The acceptance criteria are met.

The exclusion area boundary doses reported are for the worst 2 hour period, determined to be from 6 to 8 hours.

6.4 Main Steam Line Break Doses (MSLB)

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

Input Parameters and Assumptions

The analysis of the MSLB radiological consequences uses the analytical methods and assumptions outlined in the RG 1.183, Appendix E. Input parameters and assumptions are provided in Table 24.

For the pre-accident iodine spike case it is assumed that a reactor transient has occurred prior to the MSLB and has raised the RCS iodine concentration to a conservative value of 60 μCi/gm of dose equivalent (DE) I-131. (The TS 3.4.16 limit for a transient is 50 μCi/gm of dose equivalent (DE) I-131.) Table 21 documents the iodine coolant activity corresponding to 60 μCi/gm of DE I-131.

For the accident-initiated iodine spike case, the primary system transient associated with the MSLB causes an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times the appearance rate corresponding to a maximum equilibrium RCS concentration of 0.5 μCi/gm of DE I-131 (proposed TS 3.4.16). Table 22 provides the accident initiated iodine spike appearance rates used in the analysis. The spike is allowed to continue until 4 hours from the start of the event. After this point in the accident there is no activity available for release from the gap. The guidance in RG 1.183 specifies that the spike duration is 8 hours unless it can be demonstrated that the activity released by the 8 hour spike exceeds that available for release from the fuel gap from all fuel pins (Reference 2). It is assumed that "all fuel pins" refers only to those fuel pins containing cladding defects. In addition, it is assumed that once the activity associated with the defined gap fraction is released from the fuel into the primary coolant, there is no further release of activity from the fuel. There is no additional diffusion of fission products from the fuel pellets into the gap because once the reactor trips, the fuel would be cold relative to power operation.

The noble gas activity concentration in the RCS at the time the accident occurs is given in Table 6 and is based on a one percent fuel defect level. This is approximately equal to the TS 3.4.16 value of 100/E-bar μCi/gm for gross radioactivity. The iodine activity concentration of the secondary coolant at the time the MSLB occurs is assumed to be equivalent to the proposed TS 3.7.13 limit of 0.1 μCi/gm of DE I-131 and is given in Table 21.

As discussed in Section 2.2, the amount of primary-to-secondary SG tube leakage is assumed to be 0.7 gpm total (i.e., 500 gpd per SG). The leak is assumed to be distributed with 0.35 gpm to the intact steam generator and 0.35 gpm to the faulted steam generator. The density for this leakage is 47 lb_m/ft^3.

The SG connected to the broken steam line is assumed to boil dry within the initial two minutes following the MSLB. The entire liquid inventory of this SG is assumed to be steamed off and all of the iodine and alkali metal activity initially in this SG is released to the environment. In addition, all activity carried over to the faulted SG by tube leaks is assumed to be released directly to the environment with no credit taken for retention in the SG.

An iodine partition factor of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used in the intact SG.

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

For PBNP, plant cooldown to RHR operating conditions can be accomplished within 8 hours after initiation of the steam line break event. At 8 hours after the accident, the RHR system is placed into service for heat removal and there is no further steam release to the atmosphere from the intact SG. A primary and secondary side mass and energy balance was used to calculate the steam released from the intact SG from 0 to 2 hours and from 2 to 8 hours. The releases conservatively bound the assumed core power of 1683 MWt including uncertainties.

Within 30 hours after the accident, the reactor coolant system has been cooled to below 212°F, and there are no further steam releases to the atmosphere from the faulted steam generator.

Control Room Isolation

In the event of a MSLB, the low steam line pressure SI setpoint will be reached shortly after event initiation. The SI/containment isolation signal or a radiation monitor signal cause the control room HVAC to switch from the normal operation mode to the post-accident mode of operation. The analysis conservatively did not credit the SI signal but relied on the ventilation system line radiation monitor signal for control room isolation. It was confirmed that the radiation monitor setpoint is reached within 15 seconds. The control room HVAC switches from normal operation to post-accident mode of operation at 75 seconds (15 seconds for radiation signal plus 60 second delay time).

Acceptance Criteria

The EAB and LPZ dose acceptance criteria for a MSLB with a pre-accident iodine spike is 25 rem TEDE per RG 1.183. This is the 10 CFR 50.67 limit. For a MSLB with an accident initiated iodine spike, the EAB and LPZ dose acceptance criterion is 2.5 rem TEDE per RG 1.183. This is 10 percent of the 10 CFR 50.67 limit. The control room dose acceptance criterion is 5.0 rem TEDE per 10 CFR 50.67.

The EAB doses are calculated for the worst 2 hours. The LPZ doses are calculated up to the time all releases are terminated, which is the time to cool to 212°F (30 hours). The control room doses are calculated for 30 days.

Results and Conclusions

The SLB accident doses are listed below.

For the pre-accident iodine spike:

Exclusion Area Boundary	0.14 rem TEDE
Low Population Zone	0.03 rem TEDE
Control Room	2.1 rem TEDE

For the accident-initiated iodine spike:

Exclusion Area Boundary	0.19 rem TEDE
Low Population Zone	0.07 rem TEDE
Control Room	4.1 rem TEDE

The acceptance criteria are met.

The exclusion area boundary doses reported are for the worst 2 hour period, determined to be from 0 to 2 hours for the pre-accident iodine spike and from 4 to 6 hours for the accident initiated iodine spike.

6.5 Control Rod Ejection Accident Doses (CRDE)

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 clad 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 atmospheric dump valves (ADVs) or the main steam safety valves (MSSVs). Iodine and alkali metals group 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 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.

Input Parameters and Assumptions

The analysis of the rod ejection radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix H. Input parameters and assumptions are provided in Table 25.

Source Term

In determining the doses following a rod ejection accident, it is assumed that 10 percent of the fuel rods in the core suffer sufficient damage as a result of departure from nuclear boiling (DNB) and all of their gap activity is released. It is assumed that 50 percent of the rods in DNB undergo centerline melting, with the melting limited to the inner 10 percent and occurring over 50 percent of the axial length, whereby 0.25 percent of the activity in the core is released as a result of partial melting of the fuel. Ten percent of the total core activity of iodine and noble gases and 12 percent of the total core activity for alkali metals are assumed to be in the fuel cladding gap (Reference 2). Consistent with Reference 2, the activity releases from the failed/melted fuel based on the Table 5 core average activities were multiplied by the maximum radial peaking factor of 1.8.

For both the containment leakage release path and the primary-to-secondary leakage release path, all noble gas activity contained in the failed fuel gap and in the melted fuel is available for release.

For the containment leakage release path, all of the iodine and alkali metals activity contained in the failed fuel gap and 25 percent of the iodine and alkali metals activity contained in the melted fuel is available for release.

For the primary-to-secondary leakage release path all of the iodine and alkali metals activity contained in the failed fuel gap and 50 percent of the iodine and alkali metals activity contained in the melted fuel is available for release from the reactor coolant system.

Prior to the accident the iodine activity concentration of the primary coolant is 0.5 μCi/gm of DE I-131 (proposed TS 3.4.16) and is given in Table 21. The noble gas and alkali metal activity concentration in the RCS at the time the accident occurs is based on a one percent fuel defect level and is given in Table 6. This is approximately equal to the TS 3.4.16 value of 100/E-bar μCi/gm for gross radioactivity. The iodine activity concentration of the secondary

coolant at the time the rod ejection occurs is assumed to be equivalent to the proposed TS 3.7.13 limit of 0.1 μCi/gm of DE I-131 and is given in Table 21. The alkali metal activity concentration of the secondary coolant at the time the rod ejection occurs is assumed to be 20 percent of the primary side concentration.

Iodine in containment is assumed to be 4.85 percent elemental, 0.15 percent organic and 95 percent particulate.

Iodine released from the secondary system is assumed to be 97 percent elemental and 3 percent organic.

Containment Release Pathway

The containment is assumed to leak at the proposed leak rate of 0.2 percent per day (Bases to TS 3.6.1) for the first 24 hours of the accident and then to leak at half that rate (0.1 percent per day) for the remainder of the 30 day period following the accident considered in the analysis.

For the containment leakage pathway, no credit is taken for plate out onto containment surfaces or for containment spray operation, which would remove airborne particulates and elemental iodine. Sedimentation of alkali metal particulates in containment is credited. This corresponds to a decontamination factor of less than 150. 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 16). As discussed in Section 6.1, LOCA, the CSE tests determined that there was a significant sedimentation removal rate even with a relatively low aerosol concentration. From CSE, even at an air concentration of 10 $\mu g/m^3$, the sedimentation removal coefficient was above 0.3 hr^{-1}.

Primary-to-Secondary Leakage Release Pathway

When determining doses due to the primary-to-secondary SG tube leakage, all the iodine, alkali metals 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). The primary-to-secondary tube leakage continues until the RCS pressure drops below the secondary pressure. The rod ejection pressure transient is similar to that of a small break LOCA. A conservative time of 0.417 hours was used for this analysis because analyses of the small break LOCA pressure transient have shown that the primary pressure is less than the secondary pressure before this time. Steam releases from the SG are conservatively assumed to continue for 8 hours. The locked rotor steam releases are conservatively applied for this analysis. The locked rotor releases are conservative since they do not include ECCS injection to absorb decay heat.

As discussed in Section 2.2, the amount of primary-to-secondary SG tube leakage is assumed to be 0.7 gpm total (i.e., 500 gpd per SG). Although the primary-to-secondary pressure differential drops throughout the event, the constant flow rate is conservatively maintained. The density for this leakage is 47 lbm/ft^3.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) and an alkali metal partition factor in the SGs of 0.0025 (curies alkali metals/gm steam) / (curies alkali metals/gm water) is used.

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

Control Room Isolation

In the event of a rod ejection, the low pressurizer pressure SI setpoint will be reached at approximately 88.2 seconds (rounded up to 90 seconds) after event initiation. The SI/containment isolation signal cause the control room HVAC to switch from the normal operation mode to the post-accident mode of operation. It is assumed that the control room HVAC switches from normal operation to post-accident mode of operation at 150 seconds (90 seconds for SI/containment isolation signal plus a conservative 60 second delay time).

Acceptance Criteria

The exclusion area boundary and low population zone dose acceptance criteria for a rod ejection is 6.3 rem TEDE per RG 1.183. This is approximately 25 percent of the 10 CFR 50.67 limit. The control room dose acceptance criterion is 5.0 rem TEDE per 10 CFR 50.67.

The exclusion area boundary doses are calculated for the worst 2 hours. The low population zone and control room doses are calculated for 30 days.

Results and Conclusions

The rod ejection doses are:

Exclusion Area Boundary	2.0 rem TEDE
Low Population Zone	0.65 rem TEDE
Control Room	2.7 rem TEDE

The acceptance criteria are met.

The exclusion area boundary doses reported are for the worst 2 hour period, determined to be from 0 to 2 hours.

6.6 Fuel Handling Accident Doses (FHA)

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

The FHA analysis assuming an AST was previously approved in Reference 19. The current FHA assumes an emergency mode that consists of filtered makeup air only and 500 cfm of unfiltered inleakage. To remain consistent with the other non-LOCA design basis radiological accident analyses, the FHA was re-evaluated assuming a revised CR HVAC emergency mode (see Section 5.3), a reduced unfiltered inleakage (i.e., 330 cfm), and revised atmosphere dispersion factors for the control room (see Section 4.3).

In addition, higher gap fractions were applied for the fraction of the fuel that does not meet the limits on burnup and linear heat generation rate, following the method approved by the NRC for Kewaunee Power Station (ML070430020) (Reference 24). The gap fractions are those from Safety Guide 25 with the value for I-131 adjusted consistent with the recommendation in NUREG/CR-5009.

For convenience, the input assumptions are presented in Table 26.

The impact on the dose consequences to the control room are presented below. The offsite areas are not affected by these input parameters but are reproduced below for completeness.

Acceptance Criteria

The EAB and LPZ dose acceptance criterion for the FHA is 6.3 TEDE per RG 1.183. The control room dose acceptance criterion is 5.0 rem TEDE per 10 CFR 50.67.

The EAB dose is calculated for the worst 2 hours. The LPZ dose is calculated for 2 hours (i.e., the duration of the release). Control room doses are calculated for 30 days.

Results and Conclusions

The fuel handling accident doses are:

	All Rods Meet RG 1.183 Table 3 Footnote 11 Limits	No Rods Meet RG 1.183 Table 3 Footnote 11 Limits
Exclusion Area Boundary	1.6 rem	2.6 rem
Low Population Zone	0.1 rem	0.16 rem
Control Room	2.7 rem	4.2 rem

The acceptance criteria remain met.

6.7 Reactor Vessel Head Drop Accident Doses (RVHD)

The reactor vessel head is assumed to drop onto the vessel causing fuel cladding damage to all of the fuel assemblies in the core, which results in a gap release. In addition, damage to the bottom-mounted instrumentation (BMI) tubes is assumed such that approximately 300 gpm of reactor coolant is lost through these penetrations. This loss of inventory is well within the capacity of a single SI or RHR pump. Damage to the point of rupture or shearing of other connected piping, including the main RCS loops, pressurizer surge line, core deluge lines, accumulator dump lines, normal charging, and cold leg safety injection lines is not expected.

Initial makeup of the RCS to the vessel is via suction from the refueling water storage tank (RWST) to the safety injection pumps, RHR pumps or charging pumps. Once the RWST volume is exhausted, the RHR system is realigned to recirculate the coolant in the containment sump to maintain the core sub-cooled.

Input Parameters and Assumptions

The RVHD radiological analysis utilizes a combination of input assumption guidance obtained from the LOCA and FHA radiological analyses as they apply to the accident scenario. This was done because there is currently no explicit guidance in RG 1.183 that applies to this particular scenario.

The RVHD analysis assumes that containment closure is established prior to the event and the following initial conditions are assumed: 1) containment equipment hatch and personnel airlocks are closed (equipment hatch on and bolted, one access door closed in each airlock, interlocks functional); 2) purge supply/exhaust system fans are off and isolation valves closed; and 3) other containment penetrations that allow containment atmosphere to communicate with the environment or the PAB atmosphere are closed.

The event does not result in the pressurization of the containment building because the event occurs while the reactor coolant system is open to the containment building atmosphere and sufficient RCS makeup is available to provide cooling to the core. There is no release via containment leakage.

For convenience, the input assumptions are presented in Table 27.

Source Term

The accident analysis assumes that the RVHD occurs immediately upon reactor shutdown, and 100% of the fuel assemblies are damaged to produce a complete gap release. The accident occurs at temperatures and pressures well below operating levels and the accident mitigation strategy ensures that the core is covered and cooled. No additional fuel damage is assumed to occur.

The non-LOCA gap fractions are applicable to the RVHD since no additional release from the fuel due to fuel melt will occur. The amount of activity released from the gap is determined from the total core inventory assumed for the LOCA analysis with no assumed post-shutdown adjustment for decay time.

ECCS Leakage

When ECCS recirculation is established following the RVHD, leakage is assumed to occur from ECCS equipment outside containment. This leakage goes into the primary auxiliary building and is released to the environment, unfiltered, through the primary auxiliary building vent stack. Recirculation is conservatively initiated at 0 minutes. The leakage continues for the 30 day period following the accident considered in the analysis.

The amount of coolant available for recirculation is assumed to be equal to the amount of coolant that is injected (243,000 gallons of RWST inventory). There is no credit for the volume of coolant initially in the vessel or the RCS which provides a conservative concentration of iodine available for release during ECCS recirculation.

Activity enters the sump and flows out of containment in the ECCS recirculation flow and is released to the environment through leakage from the ECCS. Only iodine (100% elemental) is released through this pathway since the noble gases are not assumed to dissolve in the sump and particulates would remain in the water of the ECCS leakage. It is assumed that the iodine is instantaneously and homogeneously mixed in the containment sump water at the time of release from the core.

The total ECCS recirculation leakage into the auxiliary building modeled in the analysis is 0.2113 gpm (i.e. the assumed value of 400 cc/min total ECCS leakage outside the containment is doubled consistent with RG 1.183 guidance). Ten percent of the iodine in the leakage becomes airborne.

Control Room Isolation

In the event of a reactor vessel head drop, immediate manual actuation of the control room HVAC post-accident mode of operation is assumed to occur. There is sufficient time between accident recognition and release initiation to credit manual actuation of the control room HVAC post-accident mode of operation. In addition, the magnitude of the activity released to the environment is large enough to ensure automatic operation of the control room HVAC post-accident mode of operation via a high radiation signal when recirculation is initiated. Therefore, no delay in manual actuation of the control room HVAC post-accident mode of operation is taken into consideration for the RVHD control room dose analysis.

Acceptance Criteria

The EAB and LPZ dose acceptance criterion for the FHA is applied to the RVHD, and is 6.3 TEDE per RG 1.183. The control room dose acceptance criterion is 5.0 rem TEDE per 10 CFR 50.67.

The EAB dose is calculated for the worst 2 hours. The LPZ and control room doses are calculated for the duration of the release (i.e., 30 days).

Results and Conclusions

The reactor vessel head drop doses are:

Exclusion Area Boundary	0.3 rem TEDE
Low Population Zone	0.2 rem TEDE
Control Room	0.9 rem TEDE

The exclusion area boundary doses reported are for the worst 2 hour period, determined to be from 0 to 2 hours.

The acceptance criteria are met.

7.0 Summary of Offsite and Control Room Doses

Accident	EAB (rem)	LPZ (rem)	Offsite Dose Criteria (rem)	CR (rem)	CR Dose Criteria (rem)
Loss of Coolant Accident	13.5	1.9	25	4.8	5
SGTR Pre-Accident Spike	2.0	0.2	25	1.9	5
SGTR Accident Initiated Spike	0.6	0.1	2.5	0.5	5
Locked Rotor	2.2	0.41	2.5	3.6	5
MSLB Pre-Accident Spike	0.14	0.03	25	2.1	5
MSLB Accident Initiated Spike	0.19	0.07	2.5	4.1	5
Control Rod Ejection	2.0	0.65	6.3	2.7	5
Fuel Handling Accident[Note 1]	2.6	0.16	6.3	4.2	5
Reactor Vessel Head Drop	0.3	0.2	6.3	0.9	5

Note 1: Most conservative values are reported. See Sections 3.3 and 6.6 of this report for further information.

8.0 Conclusion

RG 1.183 (Reference 2) defines an AST model for use in evaluating the radiological consequences of a postulated large break LOCA with core melt. This AST term model also forms the basis for determining the radiological consequences for other design basis accidents as provided in RG 1.183.

This licensing amendment request provides the PBNP proposed implementation of the alternative source term methodology, as defined in RG 1.183 and its Appendices, for incorporation into the plant's DBA radiological analyses. Analyses of the radiological consequences of the large break LOCA, steam generator tube rupture, locked rotor, steam line break, rod ejection, fuel handling accident, and reactor vessel head drop accident have been made using the RG 1.183 methodology. The calculated doses do not exceed the defined acceptance criteria.

The analyses used the following key assumptions consistent with proposed changes in plant design and operation with:

- The AST methodology is adopted for the composition, magnitude, chemical form, and timing of radiation releases, as well as accident-specific modeling for all radiological design basis accidents presented in the PBNP Final Safety Analysis Report (FSAR);

- The total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11;

- Atmospheric dispersion factors for the control room intake are reanalyzed for existing pathways using ARCON96;

- New values for control room unfiltered air inleakage are assumed. The AST methodology as established in RG 1.183 is being used to calculate the offsite and control room radiological consequences for PBNP to support the control room habitability program;

- The control room ventilation system is operating with filtered recirculation in addition to a filtered fresh air intake;
- Credit is taken for future shielding modifications to the control room;
- A reduced value in the allowable dose equivalent (DE) I-131 concentrations in the primary and secondary systems is used;
- A reduced containment leakage is modeled;
- Credit is taken for the use of containment spray while on ECCS recirculation (LOCA);
- A factor of two increase is applied to the ECCS leakage limit for control room habitability radiological analysis;
- Flashing fractions are applied to the steam generator tube rupture break flow;
- Credit is not taken for the administration of potassium iodide (KI) to control room personnel;
- Elimination of the requirement that the reactor must be shut down for 100 hours prior to lifting the reactor vessel head based on the revised analyses.

In addition, this LAR provides the PBNP proposed Technical Specification changes to support implementation of AST and are as follows:

1. TS Section 1.1, Definitions. The definition of L_a used in the containment leakage rate testing program (TS 5.5.15). L_a is reduced from 0.4% of containment air weight per day at P_a to 0.2% of containment air weight per day at P_a.
2. TS Section 3.4.16, RCS Specific Activity. The specific activity of the reactor coolant is revised from DOSE EQUIVALENT I-131 >0.8 μCi/gm to >0.5 μCi/gm in the Conditions and in Surveillance Requirement SR 3.4.16.2.
3. TS Section 3.7.9, CREFS. SR 3.7.9.6 is revised for clarification of mode 5 operation.
4. TS Section 3.7.13, Secondary Specific Activity. The specific activity of the secondary coolant is revised from $\leq$1.00 to 0.1 μCi/gm DOSE EQUIVALENT I-131 in the LCO and Surveillance Requirements.
5. TS Section 5.5.15, Containment Leakage Rate Testing Program. The maximum allowable containment leakage rate, L_a at P_a shall be 0.2% of containment air weight per day.

Based on the proposed information presented in this license amendment request, it may be concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

9.0 References

1. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," US AEC, Division of Licensing and Regulation, J. J. DiNunno, et. al, March 23, 1962.

2. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

3. "Use of Alternative Source Terms at Operating Reactors," NRC Final Rule 10 CFR 50.67, issued in Federal Register, Vol. 64, No. 246, Pages 71990-72002, December 23, 1999.

4. J.V. Ramsdell, "ARCON96: Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, PNNL-10521, May 1997.

5. USNRC letter to M. Sellman, WE, "Point Beach Nuclear Plants, Units 1 and 2 – Discussion of Amendments Pertaining to Control Room Habitability (TAC Nos. MA1082 and MA1083)," April 7, 2000.

6. EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-0202, September 1988.

7. EPA Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water and Soil," EPA 402-R-93-081, September 1993.

8. ANS Standard ANSI/ANS-18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Reactors," approved December 31, 1984.

9. Murphy, K. G., Campe, K. M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," Proceedings of the Thirteenth AEC Air Cleaning Conference held August 1974, published March 1975.

10. NMC Letter, "Supplemental Response to Request for Additional Information Regarding License Amendment Request 234, Selective Scope Implementation of Alternative Source Term for Fuel Handling Accident," December 19, 2004. (ADAMS Accession ML040020027)

11. Regulatory Guide 1.4, Revison 2 "Assumptions Used For Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactor," June 1974.

12. NUREG-0800, Standard Review Plan 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 2, December 1988.

13. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003.

14. Westinghouse Nuclear Safety Advisory Letter, NSAL-00-004, "Nonconservatisms in Iodine Spiking Calculations," March 2000.

15. Atomic Industrial Forum, Industry Degraded Core Rulemaking (IDCOR) Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," December 1983.

16. WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," March 1992.

17. Letter from Robert C. Jones to Lawrence A. Walsh, "Westinghouse Owners Group-Steam Generator Tube Uncovery Issue," March 10, 1993.

18. International Commission on Radiological Protection, ICRP Publication 38 "Radionuclide Transformations; Energy and Intensity of Emissions," 1983.

19. USNRC Letter, "PBNP, Units 1 And 2 - Issuance of Amendments Re: Technical Specification 3.9.3, Containment Penetrations, Associated With Handling Of Irradiated Fuel Assemblies and Use of Selective Implementation of the Alternative Source Term for Fuel Handling Accident," April 2, 2004. (ADAMS Accession ML040680918)

20. NMC Letter, "Point Beach Nuclear Plant Dockets 50-266 and 50-301 License Amendment Request 234, Selective Scope Implementation of Alternative Source Term for Fuel Handling Accident," March 27, 2003. (ADAMS Accession ML0307970703)

21. USNRC Generic Letter 2003-01, "Control Room Habitability," June 12, 2003. (ML031620248)

22. NMC Letter, "Generic Letter 2003-01 – Supplemental Response," December 8, 2006. (ADAMS Accession No. ML063420598)

23. Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982 (reissued February 1983). (ML003740205)

24. NRC Letter to Kewaunee Power Station dated March 8, 2007. (ML070430020)

Table 1
Committed Effective Dose Equivalent Dose Conversion Factors

Isotope	DCF (rem/curie)	Isotope	DCF (rem/curie)
I-131	3.29E+04	Cs-134	4.62E+04
I-132	3.81E+02	Cs-136	7.33E+03
I-133	5.85E+03	Cs-137	3.19E+04
I-134	1.31E+02	Rb-86	6.62E+03
I-135	1.23E+03		
		Ru-103	8.95E+03
Kr-85m	N/A	Ru-105	4.55E+02
Kr-85	N/A	Ru-106	4.77E+05
Kr-87	N/A	Rh-105	9.55E+02
Kr-88	N/A	Mo-99	3.96E+03
Xe-131m	N/A	Tc-99m	3.30E+01
Xe-133m	N/A		
Xe-133	N/A	Y-90	8.44E+03
Xe-135m	N/A	Y-91	4.88E+04
Xe-135	N/A	Y-92	7.81E+02
Xe-138	N/A	Y-93	2.15E+03
		Nb-95	5.81E+03
Te-127	3.18E+02	Zr-95	2.36E+04
Te-127m	2.15E+04	Zr-97	4.33E+03
Te-129m	2.39E+04	La-140	4.85E+03
Te-129	9.00E+01	La-141	5.81E+02
Te-131m	6.40E+03	La-142	2.53E+02
Te-132	9.44E+03	Nd-147	6.84E+03
Sb-127	6.03E+03	Pr-143	8.10E+03
Sb-129	6.44E+02	Am-241	4.44E+08
		Cm-242	1.73E+07
Ce-141	8.95E+03	Cm-244	2.48E+08
Ce-143	3.39E+03		
Ce-144	3.74E+05	Sr-89	4.14E+04
Pu-238	3.92E+08	Sr-90	1.30E+06
Pu-239	4.29E+08	Sr-91	1.66E+03
Pu-240	4.29E+08	Sr-92	8.07E+02
Pu-241	8.25E+06	Ba-139	1.72E+02
Np-239	2.51E+03	Ba-140	3.74E+03

Table 2 Effective Dose Equivalent Dose Conversion Factors			
Isotope	**DCF (rem·m^3/Ci·sec)**	**Isotope**	**DCF (rem·m^3/Ci·sec)**
I-131	6.734E-2 [1]	Cs-134	0.2801
I-132	0.4144	Cs-136	0.3922
I-133	0.1088	Cs-137	0.1066 [2]
I-134	0.4810	Rb-86	1.780E-02
I-135	0.2953		
		Ru-103	8.325E-02
Kr-85m	2.768E-02	Ru-105	0.1410
Kr-85	4.403E-04	Ru-106	0.0
Kr-87	0.1524	Rh-105	1.376E-02
Kr-88	0.3774	Mo-99	2.694E-02
Xe-131m	1.439E-03	Tc-99m	2.179E-02
Xe-133m	5.069E-03		
Xe-133	5.772E-03	Y-90	7.030E-04
Xe-135m	7.548E-02	Y-91	9.620E-04
Xe-135	4.403E-02	Y-92	4.810E-02
Xe-138	0.2135	Y-93	1.776E-02
		Nb-95	0.1384
Te-127	8.954E-04	Zr-95	0.1332
Te-127m	5.439E-04	Zr-97	3.337E-02
Te-129m	5.735E-03	La-140	0.4329
Te-129	1.018E-02	La-141	8.843E-03
Te-131m	0.2594	La-142	0.5328
Te-132	3.811E-02	Nd-147	2.290E-02
Sb-127	0.1232	Pr-143	7.770E-05
Sb-129	0.2642	Am-241	3.027E-03
		Cm-242	2.105E-05
Ce-141	1.269E-02	Cm-244	1.817E-05
Ce-143	4.773E-02		
Ce-144	3.156E-03	Sr-89	2.860E-04
Pu-238	1.806E-05	Sr-90	2.786E-05
Pu-239	1.569E-05	Sr-91	0.1277
Pu-240	1.758E-05	Sr-92	0.2512
Pu-241	2.683E-07	Ba-139	8.029E-03
Np-239	2.845E-02	Ba-140	3.175E-02

Notes:

[1] Table III.1 in Federal Guidance Report 12 (Reference 7) gives 1.82E-14 Sv·m^3/Bq·sec for I-131. The conversion to rem·m^3/Ci·sec:

$$1.82E\text{-}14\ Sv\text{-}m^3/Bq\text{-}sec * 100\ rem/Sv * 3.7E10\ Bq/Ci = 6.734E\text{-}2$$

[2] This is the DCF for Ba-137m. The DCF for Cs-137 is low; however, a significant amount of Ba-137m is produced through decay. This is conservatively addressed by applying the DCF from Ba-137m to Cs-137.

Table 3[1] Offsite Breathing Rates and Atmospheric Dispersion Factors	
Time	**Offsite Breathing Rates (m^3/sec)**
0 - 8 hours	3.5E-04
8 - 24 hours	1.8E-04
>24 hours	2.3E-04
	Offsite Atmospheric Dispersion Factors (sec/m^3)
Exclusion Area Boundary[2]	5.0E-04
Low Population Zone	
0 - 8 hours	3.0E-05
8 - 24 hours	1.6E-05
1 - 4 days	4.2E-06
> 4 days	8.6E-07

Notes:
[1]No change from current licensing basis
[2]This exclusion area boundary atmospheric dispersion factor is conservatively applied during all time intervals in the determination of the limiting 2-hour period

Table 4 Control Room Parameters	
Volume	65,243 ft^3
Control Room Unfiltered In-Leakage	105 cfm LOCA 330 cfm non LOCA
Normal Ventilation Flow Rates (Mode 1) Filtered Makeup Flow Rate Filtered Recirculation Flow Rate Unfiltered Makeup Flow Rate Unfiltered Recirculation Flow Rate	 0 cfm 0 cfm 2000 cfm (Not modeled - no analyses impact)
Emergency Mode Flow Rates (Mode 5) Filtered Makeup Flow Rate Filtered Recirculation Flow Rate Unfiltered Makeup Flow Rate Unfiltered Recirculation Flow Rate	 2500 cfm 1955 cfm 0 cfm (Not modeled - no analyses impact)
Filter Efficiencies Elemental Iodine Organic (Methyl) Iodine Particulate	 95% 95% 99%
CR Radiation Monitor Sensitivity	1.0E-05 μCi/cc of Xe-133
CR Radiation Monitor Location	Control Building Roof
CR Gamma Dose Area Monitor Setpoint	2 mrem/hr
CR Gamma Dose Monitor Location	Wall in the Center of Control Room
Delay to Switch CR HVAC from Normal Operation to Post Accident Operation after receiving an isolation signal (sec)	60 seconds[(1)]
Breathing Rate - Duration of the Event	3.5E-04 m^3/sec
Occupancy Factors 0 - 24 hours 1 - 4 days 4 - 30 days	 1.0 0.6 0.4

Notes: [1] Time assumed for CR HVAC system to change configuration

Table 5[1] Core Total Fission Product Activities (Based on 102% of 1650 MWt)			
Isotope	**Activity (Ci)**	**Isotope**	**Activity (Ci)**
I-131	4.48E+07	Cs-134	9.23E+06
I-132	6.46E+07	Cs-136	2.30E+06
I-133	9.15E+07	Cs-137	5.92E+06
I-134	1.01E+08	Rb-86	9.36E+04
I-135	8.56E+07		
		Ru-103	6.84E+07
Kr-85m	1.20E+07	Ru-105	4.59E+07
Kr-85	5.45E+05	Ru-106	2.44E+07
Kr-87	2.31E+07	Rh-105	4.25E+07
Kr-88	3.25E+07	Mo-99	8.20E+07
Xe-131m	4.81E+05	Tc-99m	7.17E+07
Xe-133m	2.85E+06		
Xe-133	9.07E+07	Y-90	4.49E+06
Xe-135m	1.79E+07	Y-91	5.73E+07
Xe-135	2.33E+07	Y-92	5.93E+07
Xe-138	7.58E+07	Y-93	6.83E+07
		Nb-95	7.76E+07
		Zr-95	7.69E+07
Te-127	4.68E+06	Zr-97	7.55E+07
Te-127m	6.20E+05	La-140	8.22E+07
Te-129m	2.10E+06	La-141	7.48E+07
Te-129	1.40E+07	La-142	7.24E+07
Te-131m	6.44E+06	Nd-147	3.01E+07
Te-132	6.36E+07	Pr-143	6.84E+07
Sb-127	4.72E+06	Am-241	7.95E+03
Sb-129	1.42E+07	Cm-242	2.01E+06
		Cm-244	1.76E+05
Ce-141	7.58E+07		
Ce-143	6.96E+07	Sr-89	4.46E+07
Ce-144	5.93E+07	Sr-90	4.31E+06
Pu-238	1.58E+05	Sr-91	5.47E+07
Pu-239	1.60E+04	Sr-92	5.90E+07
Pu-240	2.33E+04	Ba-139	8.20E+07
Pu-241	6.36E+06	Ba-140	7.94E+07
Np-239	8.54E+08		

Notes

[1] See FSAR Table 14.3.5-1 for LOCA current licensing basis core activities, FSAR Table 14.1.8-4 for LR, SGTR, MSLB, CRDE current licensing basis core activities

Table 6[1] RCS Coolant Concentrations (Based on 1% Fuel Defects)	
Isotope	**Activity (μCi/gm)**
I-131	2.451
I-132	2.618
I-133	3.962
I-134	0.5429
I-135	2.049
Kr-85m	1.521
Kr-85	7.557
Kr-87	1.004
Kr-88	2.822
Xe-131m	2.723
Xe-133m	4.518
Xe-133	244.8
Xe-135m	0.4318
Xe-135	7.913
Xe-138	0.6452
Cs-134	2.109
Cs-136	2.225
Cs-137	1.726
Rb-86	0.02232

Notes:

[1] See FSAR Table 14.1.8-4 for current licensing basis non-LOCA coolant activities

Table 7 Nuclide Decay Constants			
Isotope	**Decay Constant (hr^{-1})**	**Isotope**	**Decay Constant (hr^{-1})**
I-131	0.00359	Cs-134	3.84E-05
I-132	0.301	Cs-136	2.2E-03
I-133	0.0333	Cs-137	2.64E-06
I-134	0.791	Rb-86	1.55E-03
I-135	0.105		
		Ru-103	7.35E-04
Kr-85m	0.155	Ru-105	0.156
Kr-85	7.38E-06	Ru-106	7.84E-05
Kr-87	0.545	Rh-105	1.96E-02
Kr-88	0.244	Mo-99	1.05E-02
Xe-131m	0.00243	Tc-99m	0.115
Xe-133m	0.0132		
Xe-133	0.00551	Y-90	1.08E-02
Xe-135m	2.72	Y-91	4.94E-04
Xe-135	0.0763	Y-92	0.196
Xe-138	2.93	Y-93	0.0686
		Nb-95	8.22E-04
Te-127	7.41E-02	Zr-95	4.51E-04
Te-127m	2.65E-04	Zr-97	4.1E-02
Te-129m	8.6E-04	La-140	1.72E-02
Te-129	0.598	La-141	0.176
Te-131m	2.31E-02	La-142	0.45
Te-132	8.86E-03	Nd-147	2.63E-03
Sb-127	7.5E-03	Pr-143	2.13E-03
Sb-129	0.16	Am-241	1.83E-07
		Cm-242	1.77E-04
Ce-141	8.89E-04	Cm-244	4.37E-06
Ce-143	0.021		
Ce-144	1.02E-04	Sr-89	5.72E-04
Pu-238	9.02E-07	Sr-90	2.72E-06
Pu-239	3.29E-09	Sr-91	0.073
Pu-240	1.21E-08	Sr-92	0.256
Pu-241	5.5E-06	Ba-139	0.503
Np-239	0.0123	Ba-140	2.27E-03

Table 8 Unit 2 Containment	
Input Parameter	**Value**
Meteorological Data	PB2000.met
Wind Speed Units	m/s
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	45 m
Release Type	Ground
Release Height	26.1 m
Building Area	1640 m^2
Effluent Vertical Velocity	0 m/s
Vent or Stack flow	0 m^3/s
Vent or Stack Radius	0 m
Direction to Source	270°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	32m
Control Room Air Intake Height	26.1 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/s
Surface Roughness Length	0.20 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	5.7 and 6.6

Table 9 Auxiliary Building Vent Stack (ABVS)	
Input Parameter	**Value**
Meteorological Data	PB2000.met
Wind Speed Units	m/s
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	45 m
Release Type	Ground
Release Height	43.3 m
Building Area	2046 m^2
Effluent Vertical Velocity	0 m/s
Vent or Stack flow	0 m^3/s
Vent or Stack Radius	0 m
Direction to Source	178°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	59 m
Control Room Air Intake Height	26.1 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/s
Surface Roughness Length	0.20 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0 and 0

Table 10 Unit 2 "A" MSSVs	
Input Parameter	**Value**
Meteorological Data	PB2000.met
Wind Speed Units	m/s
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	45 m
Release Type	Ground
Release Height	43.9 m
Building Area	1640 m^2
Effluent Vertical Velocity	0 m/s
Vent or Stack Flow	0 m^3/s
Vent or Stack Radius	0 m
Direction to Source	256°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	33 m
Control Room Air Intake Height	26.1 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/s
Surface Roughness Length	0.20 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0 and 0

Table 11 Unit 2 "B" MSSVs	
Input Parameter	**Value**
Meteorological Data	PB2000.met
Wind Speed Units	m/s
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	45 m
Release Type	Ground
Release Height	43.9 m
Building Area	2455 m^2
Effluent Vertical Velocity	0 m/s
Vent or Stack Flow	0 m^3/s
Vent or Stack Radius	0 m
Direction to Source	289°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	35 m
Control Room Air Intake Height	26.1 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/s
Surface Roughness Length	0.20 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0 and 0

Table 12 Unit 2 Containment Façade Penetration	
Input Parameter	**Value**
Meteorological Data	PB2000.met
Wind Speed Units	m/s
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	45 m
Release Type	Ground
Release Height	26.1 m
Building Area	1915 m^2
Effluent Vertical Velocity	0 m/s
Vent or Stack Flow	0 m^3/s
Vent or Stack Radius	0 m
Direction to Source	249°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	19 m
Control Room Air Intake Height	26.1 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/s
Surface Roughness Length	0.20 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0 and 0

Table 13 Unit 2 Purge Stack	
Input Parameter	**Value**
Meteorological Data	PB2000.met
Wind Speed Units	m/s
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	45 m
Release Type	Ground
Release Height	43.3 m
Building Area	1999 m^2
Effluent Vertical Velocity	0 m/s
Vent or Stack Flow	0 m^3/s
Vent or Stack Radius	0 m
Direction to Source	279°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	24 m
Control Room Air Intake Height	26.1 m
Terrain Elevation Difference	0 m
Minimum Wind Speed	0.5 m/s
Surface Roughness Length	0.20 m
Sector Averaging Constant	4.3
Initial Values of sigma y and sigma z	0 and 0

Table 14 Iodine Chemical Species		
Iodine Form	**RG 1.4**	**RG 1.183**
Elemental	91%	4.85%
Organic	4%	0.15%
Particulate	5%	95%

Table 15 Fission Product Release Timing		
Release Phase	**Duration (TID-14844)**	**Duration (RG 1.183)[1]**
Coolant Activity	instantaneous release	10 to 30 seconds
Gap Activity	instantaneous release	0.5 hour
Early In-vessel	instantaneous release	1.3 hour

Note 1: Releases are sequential.

Table 16 Core Fission Product Release Fractions				
	Gap Release [1]		**Early In-Vessel**	
	TID	**RG**	**TID**	**RG**
Noble gases	n/a[2]	0.05	1.0	0.95
Halogens	n/a[2]	0.05	0.5[3]	0.35
Alkali Metals	n/a	0.05	0.01[4]	0.25
Tellurium group	n/a	0	0.01[4]	0.05
Barium, Strontium	n/a	0	0.01[4]	0.02
Noble Metals (Ruthenium group)	n/a	0	0.01[4]	0.0025
Cerium group	n/a	0	0.01[4]	0.0005
Lanthanides	n/a	0	0.01[4]	0.0002

Notes:

[1] The TID-14844 methodology does not specifically address the gap release. The RG 1.183 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 100% of noble gases, with no distinction made between gap activity release and early in-vessel release. The RG 1.183 source term assumes a release of gap activity (5% of core) followed by the in-vessel release as defined.

[2] Gap fraction is not defined by TID-14844

[3] Per TID-14844, half of this is assumed to plate out instantaneously

[4] Referred to in TID-14844 as "other fission products" but not typically included in dose analyses

Table 17 RG 1.183 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	Tellurium Group	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 18[1] Assumptions Used for Large Break LOCA Dose Analysis	
Core Activity	See Table 5
Activity release fractions and timing	See Tables 15 & 16
Iodine chemical form in containment Elemental Organic (methyl) Particulate (cesium iodide)	 4.85% 0.15% 95%
Containment net free volume	1.0E+06 ft^3
Containment sprayed volume	5.82E+05 ft^3
Fan cooler units Number in operation Flow rate (per unit) Delay Time to Start	 2 33,500 cfm 90 seconds
Containment leak rates 0 – 24 hours >24 hours	 0.2 weight %/day 0.1 weight %/day
Spray Operation Injection Sprays Initiated Injection Sprays Terminated Delay Time to Recirculation Sprays Recirculation Spray Duration	 0 seconds (0.0 hours) 1.00 hours 20 minutes (0.33 hours) 4 hours
Spray flow rates Injection Recirculation	 1,111 gpm 900 gpm
Spray fall height	65.58 ft
Containment Spray Removal Coefficients Spray elemental iodine removal Injection Recirculation Spray particulate removal Injection Recirculation	 20 hr^{-1} 10 hr^{-1} 4.59 hr^{-1} 3.72 hr^{-1}
Sedimentation particulate removal (Unsprayed region: From start of event; Sprayed region: When sprays not operating.)	0.1

Notes:

[1] Current Licensing Basis information for LOCA is contained in PBNP FSAR 14.3.5

Table 18 (Cont.) Assumptions Used for Large Break LOCA Dose Analysis	
Containment Spray DF Elemental Particulate	 200 1000
Containment Sump Volume	2.43E+05 gal
Time to Initiate ECCS Recirculation[(2)]	0 min
ECCS Leak Rate to Auxiliary Building	800 cc/min
ECCS Leakage Iodine Airborne Fraction	10.0%
Atmospheric Dispersion (χ/Q) Factors Control Room, Containment Surface[(2)]: 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 1.39E-03 sec/m^3 9.80E-04 sec/m^3 3.84E-04 sec/m^3 3.46E-04 sec/m^3 3.02E-04 sec/m^3
Atmospheric Dispersion (χ/Q) Factors Control Room, Auxiliary Building Vent Stack[(3)]: 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 1.80E-03 sec/m^3 1.31E-03 sec/m^3 5.15E-04 sec/m^3 4.03E-04 sec/m^3 3.03E-04 sec/m^3

Notes:

[2] Used for activity released via containment leakage
[3] Used for activity released via ECCS leakage

Table 19[1] Assumptions Used for SGTR Dose Analysis	
Source Data	
Reactor Coolant Iodine Activity (Initial) Pre-Accident Spike Accident-Initiated Spike	 60 μCi/gm of DE I-131 - See Table 21. 0.5 μCi/gm of DE I-131 - See Table 21. The iodine appearance rates assumed for the accident-initiated spike presented in Table 22. Spike duration is 8 hours.
Noble Gas Activity	Primary coolant noble gas activities based on 1 percent fuel defects presented in Table 6.
Secondary System Initial Activity Iodine Noble Gas	 0.1 μCi/gm DE I-131 – See Table 21. (Noble gases are not contained in the secondary system.)
Reactor Coolant Initial Mass	1.07E+08 grams
Steam Generator Initial Mass (each)	3.19E+07 grams
Offsite power	Lost at time of reactor trip (t =147.74 sec)
Primary-to-Secondary Leakage Duration for intact SG	8 hours
Iodine Species Released to Atmosphere Elemental Organic Particulate (cesium iodide)	 97% 3% 0%

Notes:

[1] Current Licensing Basis information for SGTR is contained in PBNP FSAR 14.2.4

Table 19 (Cont.) Assumptions Used for SGTR Dose Analysis		
Activity Release Data		
Ruptured Steam Generator		
Pre-trip Rupture Break Flow	26,165 lbm	(0 - 2.46 min)
Post-trip Rupture Break Flow	97,435 lbm	(2.46 - 30 min)
Pre-trip Break Flow Flash Fraction	19.53 %	(0 –2.46 min)
Post Trip Break Flow Flash Fraction	12.85 %	(2.46 – 30 min)
Steam Release	74,000 lbm	(0 – 30 min)
SG Iodine Partition Factor:		
Non-flashed	0.01	
Flashed	1.0	
Intact Steam Generator		
Primary-to-Secondary Leakage	0.35 gpm	
Steam Release	232,600 lbm	(0 – 2 hr)
	6.238E5 lbm	(2 – 8hr)
Feedwater Flow	190,800 lbm	(0 – 2 hr)
	1,392,000 lbm	(2 – 24hr)
SG Iodine Partition Factor	0.01	
Condenser Iodine partition factor	0.01	
Atmospheric Dispersion (χ/Q) Factors		
Control Room, Unit 2 "A" Safeties:		
0 – 2 hours	4.66E-03 sec/m^3	
2 – 8 hours	3.40E-03 sec/m^3	
8 – 24 hours	1.17E-03 sec/m^3	
24 – 96 hours	1.07E-03 sec/m^3	
96 – 720 hours	9.05E-04 sec/m^3	

Table 20 Thyroid Dose Conversion Factors		
Nuclide	**Current DCF Value (rem/curie) (FSAR Tables 14.1.8-3 and 14.3.5-2)**	**AST DCF Value (Sv/Bq) (Reference 6)**
I-131	1.07E+6	2.92E-07
I-132	6.29E+3	1.74E-09
I-133	1.81E+5	4.86E-08
I-134	1.07E+3	2.88E-10
I-135	3.14E+4	8.46E-09

Table 21 Iodine Specific Activities (μCi/gm)[1] Primary Coolant Based on 0.5 and 60.0 μCi/gm of DE I-131 Secondary Coolant Based on 0.1 μCi/gm of DE I-131			
	Primary Coolant		**Secondary Coolant**
Nuclide	**0.5 μCi/gm**	**60 μCi/gm**	**0.1 μCi/gm**
I-131	0.3846	46.15	0.07692
I-132	0.4108	49.30	0.08216
I-133	0.6216	74.59	0.12432
I-134	0.0852	10.22	0.01704
I-135	0.3215	38.58	0.06430

Note:
[1] From Reference 15

Table 22 Iodine Spike Appearance Rates (Ci/min)[1] Based on 0.5 μCi/gm of DE I-131 Primary Coolant Activity					
Primary Activity	**I-131**	**I-132**	**I-133**	**I-134**	**I-135**
335 times the equilibrium rate (SGTR)	54.5	142.7	100.9	58.2	68.1
500 times the equilibrium rate (MSLB)	81.4	213.0	150.6	86.9	101.6

Note:
[1] Assumptions from Reference 14 applied

Table 23[1] Assumptions Used for Locked Rotor Dose Analysis	
Source Term	
Core Activity	See Table 5
Fraction of fuel rods in core assumed to fail for dose considerations	100% of core
Gap Fractions I-131 Kr-85 Other Iodines and Noble Gases Alkali Metals	 8% of core activity 10% of core activity 5% of core activity 12% of core activity
Iodine Form (Atmospheric Release) Elemental Organic Particulate (cesium iodide)	 97% 3% 0%
Reactor Coolant Activity (Initial) Iodine Noble Gas Alkali Metal	 0.5 μCi/gm of DE I-131 – See Table 21 1.0% fuel defect level – See Table 6 1.0% fuel defect level – See Table 6
Secondary Coolant Activity (Initial) Iodine	 0.1 μCi/gm of DE I-131 – See Table 21
Release Modeling	
Primary-to-Secondary Leakage	0.70 gpm total
Steam Release to Environment 0 – 2 hours 2 – 8 hours	 204,000 lbm 443,000 lbm
SG Iodine Water/Steam Partition Coefficient	0.01
SG Alkali Metal Water/Steam Partition Coefficient	0.0025
RCS mass	1.07E8 gm
Secondary Side mass 0 – 2 hours > 2 hours	 3.19E+07 gm/SG 3.69E+07 gm/SG
Atmospheric Dispersion (χ/Q) Factors Control Room, Unit 2 "A" Safeties: 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 4.66E-03 sec/m^3 3.40E-03 sec/m^3 1.17E-03 sec/m^3 1.07E-03 sec/m^3 9.05E-04 sec/m^3

Notes:

[1] Current Licensing Basis information for LR is contained in PBNP FSAR 14.1.8

Table 24[1] Assumptions Used for Steam Line Break Dose Analysis	
Source Term	
Reactor Coolant Activity (Initial) Pre-Accident Iodine Spike Accident-Initiated Iodine Spike Noble Gas	 60 μCi/gm of DE I-131 - See Table 21 0.5 μCi/gm of DE I-131 – See Table21 1.0% fuel defect level – See Table 6
Reactor Coolant Accident-Initiated Iodine Appearance Rate Spike Factor	500 times equilibrium rate – See Table 22
Duration of Accident-Initiated Iodine Spike	4 hrs
Secondary Coolant Activity (Initial) Iodine	 0.1μCi/gm of DE I-131 – See Table 21
Release Modeling	
Primary-to-Secondary Leakage	0.7 gpm total
Steam Release from Faulted SG	5.7E+07 gm
Time to Release Initial Mass in Faulted SG	2 min
Time to Cool RCS Below 212°F (Releases from Faulted SG)	30 hrs
Iodine Form (Atmospheric Release) Elemental Organic Particulate (cesium iodide)	 97% 3% 0%
SG Tube Leak Rate	0.70 gpm total (0.35 gpm/SG)
Steam Releases to Environment 0 – 2 hours 2 – 8 hours	 213,000 lbm 413,000 lbm

Notes:
[1] Current Licensing Basis information for MSLB is contained in PBNP FSAR 14.2.5

Table 24 (Cont.) Assumptions Used for Steam Line Break Dose Analysis	
SG Iodine Water/Steam Partition Coefficient Faulted SG Intact SGs	 1.0 0.01
RCS Mass	1.07E+08 gm
Initial Faulted Secondary SG Side mass	5.7E+07 gm
Initial Intact SG mass	3.19E+07 gm
Atmospheric Dispersion (χ/Q) Factors Control Room, Unit 2 "A" Safeties[(2)]: 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 4.66E-03 sec/m^3 3.40E-03 sec/m^3 1.17E-03 sec/m^3 1.07E-03 sec/m^3 9.05E-04 sec/m^3
Atmospheric Dispersion (χ/Q) Factors Control Room, Unit 2 Containment Façade Penetration[(3)]: 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 1.87E-02 sec/m^3 1.50E-02 sec/m^3 5.11E-03 sec/m^3 4.94E-03 sec/m^3 4.23E-03 sec/m^3

Notes:
[2] Used for activity released via the *intact* steam generator
[3] Used for activity released via the *faulted* steam generator

Table 25[1] Assumptions Used for Rod Ejection Dose Analysis	
Source Term	
Core Activity	See Table 5
Fraction of Fuel Rods in Core that Fail	10 % of core
Gap Fractions Iodine Noble Gas Alkali Metals	 10% of core activity 10% of core activity 12% of core activity
Fraction of Fuel Melting	0.25% of core
Radial Peaking Factor	1.8
Fraction of Activity Released from Failed Fuel (Gap Activity) Containment leakage Primary-to-Secondary leakage	 100% 100%
Fraction of Activity Released from Melted Fuel Containment leakage Iodine Noble Gas Alkali Metals Primary-to-Secondary leakage Iodine Noble Gas Alkali Metals	 25% 100% 25% 50% 100% 50%
Reactor Coolant Activity (Initial) Iodine Noble Gas Alkali Metal	 0.5 μCi/gm DE I-131 – See Table 21 1.0% fuel defect level – See Table 6 1.0% fuel defect level – See Table 6
Secondary Coolant Activity (Initial) Iodine Alkali Metal	 0.1 μCi/gm DE I-131 - See Table 21 20% of primary concentration

Notes:

[1] Current Licensing Basis information for CRDE is contained in PBNP FSAR 14.2.6

Table 25 (Cont.) Assumptions Used for Rod Ejection Dose Analysis	
Containment Leakage Release Path	
Containment net free volume	1.0E+06 ft^3
Containment leak rates 0 – 24 hours > 24 hours	 0.2 weight %/day 0.1 weight %/day
Iodine chemical form in containment Elemental Organic Particulate (cesium iodide)	 4.85% 0.15% 95%
Spray removal in containment	Not Credited
Sedimentation removal in containment Iodines Alkali metals	 Not Credited 0.1 hr^{-1}
Primary-to-Secondary Leakage Release Path	
Primary-to-Secondary Leakage	0.7 gpm total
Steam release to environment 0 – 2 hours 2 – 8 hours	 204,000 lbm 443,000 lbm
SG iodine water/steam partition coefficient	0.01
SG alkali metal water/steam partition coefficient	0.0025
Iodine chemical form in after release to atmosphere Elemental Organic Particulate (cesium iodide)	 97% 3% 0%
RCS mass	1.07E+08 gm
Intact SG mass	3.19E+07 gm

Table 25 (Cont.) Assumptions Used for Rod Ejection Dose Analysis	
Primary-to-Secondary Leakage Release Path (Cont.)	
Atmospheric Dispersion (χ/Q) Factors Control Room, U2 Containment Surface[(2)]:	
0 – 2 hours	1.39E-03 sec/m^3
2 – 8 hours	9.80E-04 sec/m^3
8 – 24 hours	3.84E-04 sec/m^3
24 – 96 hours	3.46E-04 sec/m^3
96 – 720 hours	3.02E-04 sec/m^3
Atmospheric Dispersion (χ/Q) Factors Control Room, Unit 2 "A" Safeties[(3)]:	
0 – 2 hours	4.66E-03 sec/m^3
2 – 8 hours	3.40E-03 sec/m^3
8 – 24 hours	1.17E-03 sec/m^3
24 – 96 hours	1.07E-03 sec/m^3
96 – 720 hours	9.05E-04 sec/m^3

Notes:
[2] Used for activity released via containment leakage
[3] Used for activity released via secondary releases

Table 26[1] Assumptions Used for FHA in Containment Dose Analysis	
Radial Peaking Factor	1.8
Fuel Damaged	1 assembly
Time from Shutdown before Fuel Movement	65 hrs
Activity in the Damaged Fuel Assembly I-131 I-132 I-133 I-135 Kr-85m Kr-85 Kr-88 Xe-131m Xe-133m Xe-133 Xe-135m Xe-135	 3.00E+05 Ci 3.05E+05 Ci 8.87E+04 Ci 7.81E+02 Ci 4.31E+00 Ci 4.50E+03 Ci 3.45E-02 Ci 3.93E+03 Ci 1.45E+04 Ci 6.17E+05 Ci 1.25E+02 Ci 1.26E+04 Ci
Iodine chemical form in pool Elemental Organic (methyl) Particulate (cesium iodide)	 99.85% 0.15% 0%
Gap Fractions I-131 Kr-85 Other Iodines and Noble Gases	 12% of core activity 30% of core activity 10% of core activity
Water Depth	23 feet
Overall Pool Iodine Scrubbing Factor	200
Filter Efficiency	No filtration assumed
Isolation of Release	No isolation assumed
Time to Release All Activity	2 hrs
Atmospheric Dispersion (χ/Q) Factors Control Room, Unit 2 Purge Stack[2]: 0 – 2 hours	 6.94E-03 sec/m^3

Note:

[1] Current Licensing Basis information for FHA is contained in PBNP FSAR 14.2.1

[2] The Unit 2 purge stack is the bounding release point for an accident in either the containment or auxiliary building (spent fuel pool)

Table 27[1] Assumptions Used for RVHD Dose Analysis	
Core Total Fission Product Activity	See Table 5
Fuel Damaged	100%
Fuel Melt	0%
Time from Shutdown before Head Movement	Immediate
Gap Fraction – Iodine	5%
Iodine Form Released to Environment Elemental Organic	 100% 0%
Recirculation Initiation Time	Immediate
Iodine Airborne Fraction	10.0%
Containment Sump Volume	2.43E+05 gal
Control Room Isolation	Immediate Manual Operator Action
Atmospheric Dispersion (χ/Q) Factors Control Room, Auxiliary Building Vent Stack[1]: 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 1.80E-03 sec/m^3 1.31E-03 sec/m^3 5.15E-04 sec/m^3 4.03E-04 sec/m^3 3.03E-04 sec/m^3
ECCS Leak Rate to Auxiliary Building	800 cc/min

Note:

[1] Current Licensing Basis information for RVHD is contained in PBNP FSAR 14.3.6

Figure 1
Unit 1, Unit 2 and Common Release Locations

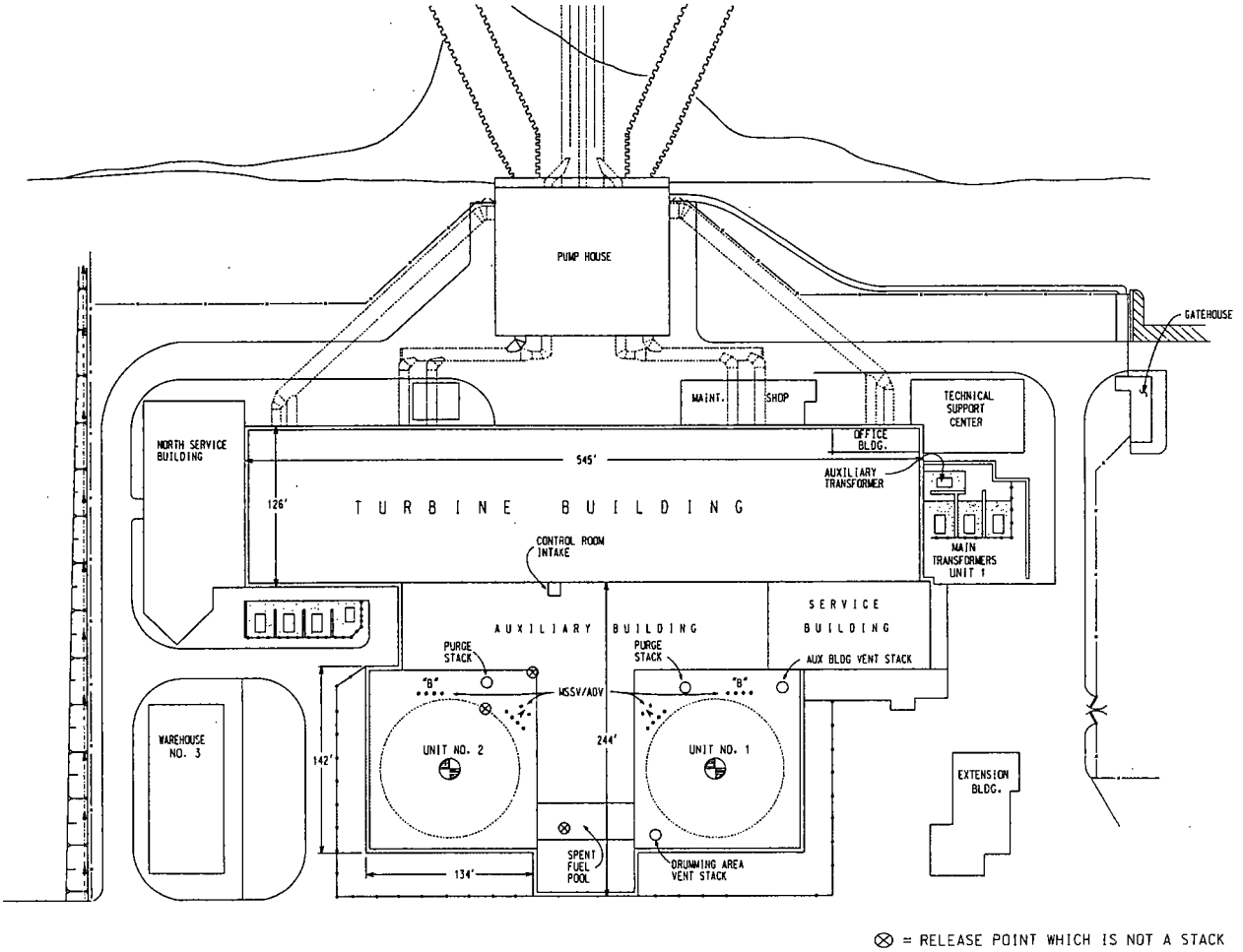
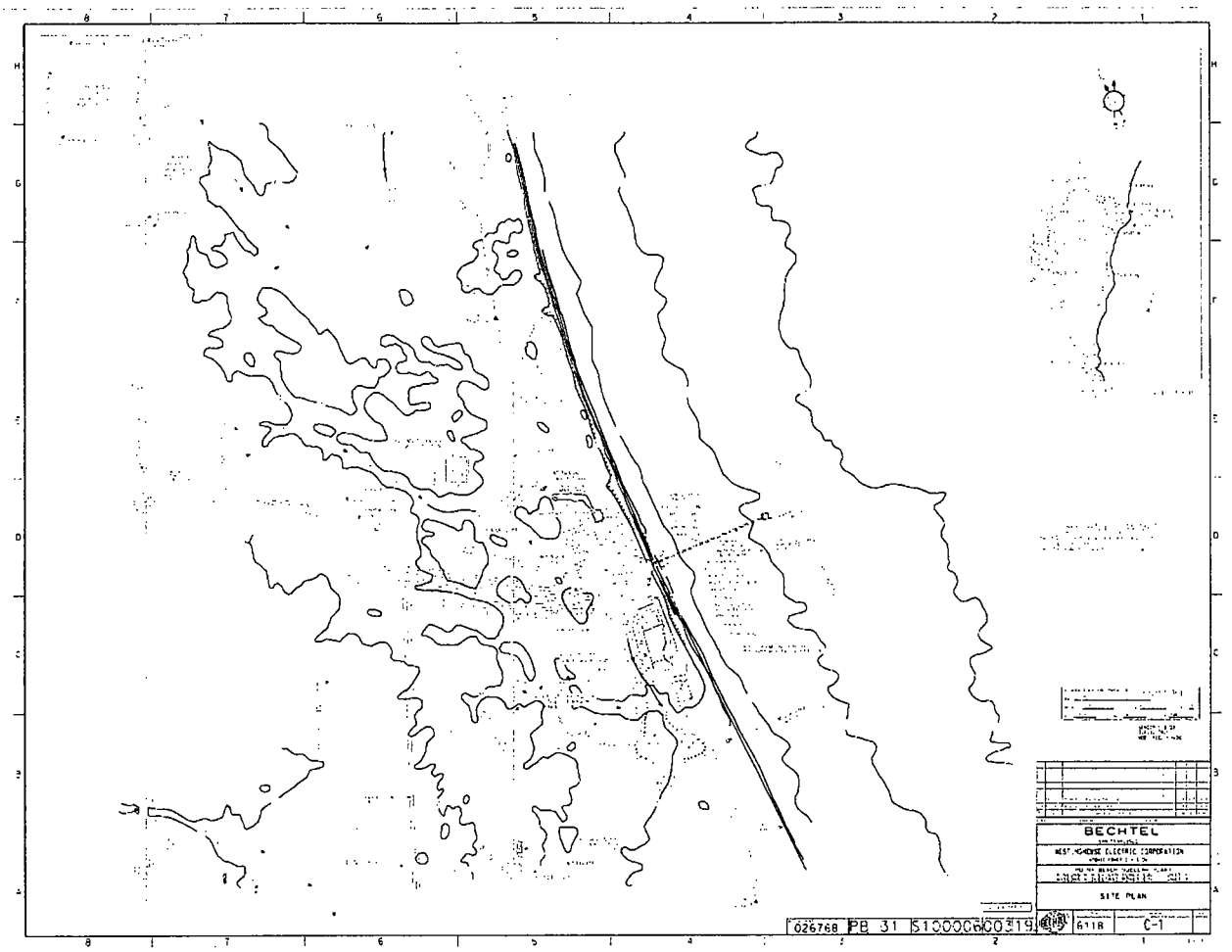


Figure 2
Point Beach Nuclear Plant Site Plan



ENCLOSURE 4

LICENSE AMENDMENT REQUEST 241
REGULATORY ISSUES SUMMARY 2006-04 RESOLUTION MATRIX

POINT BEACH NUCLEAR PLANT

5 Pages Follow

RIS 2006-04 ISSUE	ADDRESSED BY
1. Level of Detail Contained in LARs An AST amendment request should describe the licensee's analyses of the radiological and non-radiological impacts and provide a justification for the proposed modification in sufficient detail to support review by the NRC staff. For example, the AST amendment request should (1) provide justification for each individual proposed change to the technical specifications (TS), (2) identify and justify each change to the licensing basis accident analyses, and (3) contain enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations.	Provided in the AST License Amendment Request and the AST Technical Evaluation, Enclosure 3. Specific details on the individual radiological analyses are provided in Section 6 of the Enclosure 3.
2. Main Steam Isolation Valve (MSIV) Leakage and Fission Product Deposition in Piping For calculation of aerosol settling velocity in the main steam line (MSL) piping of boiling water reactors, some LARs reference Accident Evaluation Report (AEB) 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term" (Ref. 2). This is acceptable. However, it is important to note that the report was written based on the parameters of a particular plant and, therefore, the removal rate constant is specific to that plant. Any licensee who chooses to reference these AEB 98-03 assumptions should provide appropriate justification that the assumptions are applicable to their particular design.	Not Applicable. Applicable only to BWRs.
3. Control Room Habitability When implementing an AST, some licensees have proposed that certain engineered safety features (ESF) ventilation systems not be credited as a mitigation feature in response to an accident. In some cases, the licensee's revised design basis analysis introduced the assumption that normal (non-ESF) ventilation systems are operating during all or part of an accident scenario. Such an assumption is inappropriate unless the non-ESF system meets certain qualities, attributes, and performance criteria as described in RG 1.183, Regulatory Positions 4.2.4 and 5.1.2. For example, credit for the operation of non-ESF ventilation systems should not be assumed unless they have a source of emergency power. In addition, the operation of ventilation systems establishes certain building or area pressures based upon their flow rates. These pressures affect leakage and infiltration rates which ultimately affect operator dose. Therefore, to credit the use of these systems, licensees should incorporate the systems into the ventilation filter testing program in Section 5 of the TS. In summary, use of non-ESF ventilation systems during a DBA should not be assumed unless the systems have emergency power and are	No credit is taken for non-ESF ventilation systems during a DBA.

RIS 2006-04 ISSUE	ADDRESSED BY
part of the ventilation filter testing program in Section 5 of the TS. Generic Letter (GL) 2003-01, "Control Room Habitability" (Ref. 5) requested licensees to confirm the ability of their facility's control room to meet applicable habitability regulatory requirements. In addition, licensees were requested to confirm that control room habitability systems were designed, constructed, configured, operated and maintained in accordance with the facility's design and licensing bases.	PBNP's response to GL 2003-01 indicated that the unfiltered inleakage assumption in the control room habitability analyses was non-conservative. For mode 4 operation of the control room emergency ventilation system, the measured value of 96 cfm using tracer gas testing exceeded the assumed 10 cfm unfiltered inleakage analysis input assumption (ML063420598). It is anticipated that modification of the control room emergency ventilation system to allow a combination of outside air and return air to pass through the emergency HEPA/charcoal filter unit to the suction of the control room recirculation fan (new mode 5 operation) will result in unfiltered inleakage value less than the 105 cfm unfiltered inleakage assumption associated with the LOCA radiological analysis. This will allow control room habitability systems to be configured, operated and maintained in accordance with the new facility design and licensing bases. The modification to the control room emergency ventilation system to allow a combination of outside air and return air to pass through the emergency HEPA/charcoal filter unit to the suction of the control room recirculation fan (new Mode 5 operation) is in development.
4. Atmospheric Dispersion Licensees may continue to use atmospheric relative concentration (χ/Q) values and methodologies from their existing licensing-basis analyses when appropriate. Licensees also have the option to adopt the generally less conservative (more realistic) updated NRC staff guidance on determining χ/Q values in support of design basis control room radiological habitability assessments provided in RG 1.194, "Atmospheric Relative Concentrations for Control	The PBNP AST license amendment request includes revised atmospheric relative concentration values for onsite radiological consequence analyses developed in conformance with RG 1.194 as described in Section 4 of the AST Technical Evaluation, Enclosure 3.

RIS 2006-04 ISSUE	ADDRESSED BY
Room Radiological Habitability Assessments at Nuclear Power Plants" (Ref. 6). Regulatory positions on χ/Q values for offsite (i.e., exclusion area boundary and low population zone) accident radiological consequence assessments are provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (Ref. 7). Based on submittal reviews, the NRC staff identified the following areas of improvement for licensee submittals that propose revision of the design basis atmospheric dispersion analyses for implementing AST. They should include the following information:	Offsite radiological consequence analyses utilized current licensing basis atmospheric relative concentration values.
• A site plan showing true north and indicating locations of all potential accident release pathways and control room intake and unfiltered inleakage pathways (whether assumed or identified during inleakage testing).	A site plan indicating true north and locations of releases and receptor points is included in Enclosure 3 (Figures 1 and 2).
• Justification for using control room intake χ /Q values for modeling the unfiltered inleakage, if applicable.	Justification for the atmospheric relative concentrations used for the control room envelope unfiltered inleakage is provided in Enclosure 3 Section 4.3.
• A copy of the meteorological data inputs and program outputs along with a discussion of assumptions and potential deviations from staff guidelines. Meteorological data input files should be checked to ensure quality (e.g., compared against historical or other data and against the raw data to ensure that the electronic file has been properly formatted, any unit conversions are correct, and invalid data are properly identified).	Meteorological data and drawings are enclosed in Enclosure 6. Assumptions and atmospheric relative concentrations are included in Enclosure 3. The electronic data has been verified to be properly converted and formatted.
5. Modeling of ESF Leakage ESF systems that recirculate sump water outside the primary containment may leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (e.g., refueling water storage tank). Appendix A to RG 1.183, Regulatory Position 5, states that "the radiological consequences from the postulated [ESF] leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the [loss-of-coolant accident] LOCA."	The postulated ESF leakage is analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The total ECCS recirculation leakage modeled in the PBNP LOCA analysis is twice the maximum allowable FSAR value and is consistent with RG 1.183 guidance. The assumed amount of iodine that may become airborne from ESF leakage is 10%.
6. Release Pathways Changes to the plant configuration associated with an LAR (e.g., an "open" containment during refueling) may require a re-analysis of the design basis dose calculations. A request	Consistent with current licensing basis, the radiological analysis for Reactor Vessel Head Drop accident assumes

RIS 2006-04 ISSUE	ADDRESSED BY
for TS modifications allowing containment penetrations (i.e., personnel air lock, equipment hatch) to be open during refueling cannot rely on the current dose analysis if this analysis has not already considered these release pathways. RG 1.194, Regulatory Position 3.2.4.2 supports review of penetration pathways, by stating that "leakage is more likely to occur at a penetration, [and that the] analysts must consider the potential impact of leakage from building penetrations exposed to the environment." Therefore, releases from personnel air locks and equipment hatches exposed to the environment and containment purge releases prior to containment isolation need to be addressed.	containment closure conditions.
7. Primary to Secondary Leakage Some analysis parameters can be affected by density changes that occur in the process steam. The NRC staff continues to find errors in LAR submittals concerning the modeling of primary to secondary leakage during a postulated accident. This issue is discussed in Information Notice (IN) 88-31, "Steam Generator Tube Rupture Analysis Deficiency," (Ref. 11) and Item 3.f in RIS 2001-19. An acceptable methodology for modeling this leakage is provided in Appendix F to RG 1.183, Regulatory Position 5.2.	The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of the surveillance tests and is described in the accident specific descriptions of Section 6.2 of Enclosure 3.
8. Elemental Iodine Decontamination Factor (DF) Appendix B to RG 1.183 provides assumptions for evaluating the radiological consequences of a fuel handling accident. If the water depth above the damaged fuel is 23 feet or greater, Regulatory Position 2 states that "the decontamination factors for the elemental and organic [iodine] species are 500 and 1, respectively, giving an overall effective decontamination factor of 200." However, an overall DF of 200 is achieved when the DF for elemental iodine is 285, not 500.	There is no change to the DF for the Fuel Handling Accident from previously licensed analysis using AST.
9. Isotopes Used in Dose Assessments For some accidents (e.g., main steam line break and rod drop), licensees have excluded noble gas and cesium isotopes from the dose assessment. The inclusion of these isotopes should be addressed in the dose assessments for AST implementation.	Noble gas and cesium isotopes were considered in the dose assessments.
10. Definition of Dose Equivalent 131 In the conversion to an AST, licensees have proposed a modification to the TS definition of dose equivalent I-131. Some have modified the definition to base it upon the thyroid dose conversion factors of International Commission on Radiation Protection (ICRP) Publication 2, "Report of Committee II on Permissible Dose for Internal Radiation" (Ref. 12) or ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 13). Others have proposed a definition which is a combination of different	The TS definition of dose equivalent I-131 has not been modified.

RIS 2006-04 ISSUE	ADDRESSED BY
iodine dose conversion factors, (e.g., RG 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR [Part] 50, Appendix I" (Ref. 14), ICRP Publication 2, Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 15). Although different references are available for dose conversion factors, the TS definition should be based on the same dose conversion factors that are used in the determination of the reactor coolant dose equivalent iodine curie content for the main steam line break and steam generator tube rupture accident analyses.	
11. Acceptance Criteria for Off-Gas or Waste Gas System Release As part of full AST implementation, some licensees have included an accident involving a release from their off-gas or waste gas system. For this accident, they have proposed acceptance criteria of 500 millirem (mrem) total effective dose equivalent (TEDE). The acceptance criterion for this event is that associated with the dose to an individual member of the public as described in 10 CFR Part 20, "Standards for Protection Against Radiation." When the NRC revised 10 CFR Part 20 to incorporate a TEDE dose, the offsite dose to an individual member of the public was changed from 500 mrem whole body to 100 mrem TEDE. Therefore, any licensee who chooses to implement AST for an off-gas or waste gas system release should base its acceptance criteria on 100 mrem TEDE. Licensees may also choose not to implement AST for this accident and continue with their existing analysis and acceptance criteria of 500 mrem whole body.	The LOCA radiological analysis bounds the Volume Control Tank Rupture, Waste Gas Decay Tank Rupture and Charcoal Decay Tank Failure radiological dose analyses. These FSAR Chapter 14 radiological analyses are being relocated to system-specific FSAR sections, outside of the Chapter 14 accident analyses. These accidents are not included in the AST license amendment request.
12. Containment Spray Mixing Some plants with mechanical means for mixing containment air have assumed that the containment fans intake air solely from a sprayed area and discharge it solely to an unsprayed region or vice versa. Without additional analysis, test measurements or further justification, it should be assumed that the intake of air by containment ventilation systems is supplied proportionally to the sprayed and unsprayed volumes in containment.	The containment building is modeled as two discrete volumes: sprayed and unsprayed. The volumes are conservatively assumed to be mixed only by the containment fan coolers. The containment volume is 1.0E6 ft^3 with a sprayed fraction of 58.2 percent of the total (5.82E5 ft^3).

ENCLOSURE 5

LICENSE AMENDMENT REQUEST 241
REGULATORY GUIDE 1.183 COMPLIANCE MATRIX

POINT BEACH NUCLEAR PLANT

30 Pages Follow

REGULATORY GUIDE 1.183 COMPLIANCE MATRIX

Notes:
1. Reference to Tables or Sections in this column refers to License Amendment Request 241, Enclosure 3 unless otherwise noted.
2. RG 1.183 does not contain guidance for RVHD. Applicable LOCA and FHA guidance has been used. See "Enclosure" 3, Section 6.7.

Regulatory Guide 1.183 Main Body			
RG Section	**Regulatory Position**	**Analysis**	**Comments**
3. – ACCIDENT SOURCE TERM			
3.1 – Fission Product Inventory	The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. The uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP. For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.	Conforms	The source terms derived from the reactor core fission product inventory are listed in Table 5 of Enclosure 3. The inventory of the fission products in the reactor core is based on maximum full-power operation of the core at a power level equal to 1683 MWt, and current licensed values of fuel enrichment and burnup. The core mass calculated is 48.0 MTU with an equilibrium cycle length of 17,175 MWD/MTU. The core and coolant activities in Tables 5 and 6 of Enclosure 3 are based on a core power of 1650 MWt increased to 1683 MWt to include a 2 percent power uncertainty, which bounds the currently licensed power level of 1540 MWt plus Appendix K uncertainty. No shutdowns are modeled between cycles, while strictly conservative, this simplification is expected to have virtually no effect on core inventory. This is sufficient to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. For the equilibrium cycle, fuel burnup and fission product values were modeled via the ORIGEN 2 code. The ORIGEN 2 run models a single assembly in each of seven regions. Burnup calculations reflecting each of the appropriate power histories are performed. The total inventory for each region at the end of the equilibrium cycle is then determined by multiplying the assembly value by the number of assemblies per region. The seven

	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.		regions are summed to produce a total core inventory. For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory is being used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core, multiplying by the number of damaged rods, and multiplying by the radial peaking factor of 1.8 to determine the inventory of the damaged rods. No adjustment to the fission product inventory is made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. The FHA assumes a decay time of 65 hours, and the RVHD is assumed to occur at shutdown.
3.2 – Release Fractions	The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor.	Conforms	The assumed core inventory release fractions for each radionuclide group for both the gap release and early in-vessel damage phases for DBA LOCAs are those of Table 16 of Enclosure 3 and conform to Table 2 of RG 1.183. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. For non-LOCA events, see discussion provided in Enclosure 3, Section 3.3.
s3.3 – Timing of Release Phases	The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase. For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.	Conforms	The activity released from the core during each release phase is modeled in accordance with RG 1.183 Position 3.3. The leak-before-break methodology is not credited in the AST analyses.

	For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.		
3.4 – Radionuclide Composition	Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses.	Conforms	The elements assumed in each radionuclide group are consistent with those of Table 5 of RG 1.183.
3.5 – Chemical Form	Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95% of the iodine released should be assumed to be cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	Conforms	The assumed chemical form of iodine released to containment is 95% particulate, 4.85% elemental, and 0.15% organic. This applies to the LOCA and CRDE containment releases. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form. For the LOCA DBA, the assumed chemical form of iodine released from the ECCS is 97% elemental and 3% organic. For the MSLB, SGTR, LR, CRDE (steam generator release only), the assumed chemical form of iodine released is 97% elemental and 3% organic. For the FHA, the assumed chemical form of iodine in the water pool is 99.85% elemental and 0.15% organic. For the RVHD, the assumed chemical form of iodine released is 100% elemental.
3.6 – Fuel Damage in Non-LOCA DBAs	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has	Conforms	The non-LOCA design bases analyses use CLB values for the fuel damage criterion. These values are provided in PBNP FSAR Chapter 14.

	traditionally relied upon the departure from nucleate 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.		
4. – DOSE CALCULATIONAL METHODOLOGY			
4.1 – Offsite Dose Consequences	The following assumptions should be used in determining the TEDE for persons located at or beyond the boundary of the exclusion area (EAB).		
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.	Conforms	The dose calculations determine the TEDE dose, with all significant progeny included, as the sum of the CEDE and the DDE.
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff.	Conforms	The dose conversion factors (DCF's) used in determining the CEDE dose are from EPA Federal Guidance Report No. 11 and are provided in Enclosure 3 Table 1.
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5 x 10-4 cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8 x 10-4 cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3 x 10-4 cubic meters per second.	Conforms	The assumed offsite breathing rates are those specified in Section 4.1.3 of RG 1.183.

4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE.	Conforms	The EDE are used to determine the submergence dose in a semi-infinite cloud. The assumed conversion factors are those of Federal Guidance Report 12 Table 2 Enclosure 3.
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release.	Conforms	The TEDE are determined for the most limiting person for a two-hour period at the EAB and the maximum two-hour dose is reported.
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms	The TEDE is determined for the most limiting person at the LPZ.
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No plume depletion due to ground deposition is credited.
4.2 – Control Room Dose Consequences	The following guidance should be used in determining the TEDE for persons located in the control room.		
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel.	Conforms	The radiation dose to personnel within the CRE includes inhalation and immersion doses due to releases from containment and the auxiliary building, and includes direct shine doses from contained sources and the external plume.
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.	Conforms	The control room doses are determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, resulting in conservative results for the control room.
4.2.3	The models used to transport radioactive material into and through the control room, and the shielding models used to	Conforms	The models used to transport radioactive material into and through the control room, and the shielding

	determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.		models used to determine radiation dose rates from external sources, are structured to provide suitably conservative estimates of the exposure to control room personnel.
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance.	Conforms	Credit is taken for CR emergency intake and recirculation filtration. The credited filters are qualified and acceptable per the PBNP Ventilation Filter Testing Program (TS 5.5.10).
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs.	Conforms	No credit is taken for the use of personal protective equipment or prophylactic drugs.
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5 x 10-4 cubic meters per second.	Conforms	The assumed breathing rates and occupancy factors for control room operator dose are those specified in Section 4.2.6 of RG 1.183.
4.2.7	Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses.	Conforms	Control room doses are calculated using dose conversion factors identified in Regulatory Position 4.1 above. The equation given in RG 1.183 for Regulatory Position 4.2.7 is utilized for finite cloud correction when calculating immersion doses due to the airborne activity inside the control room.
4.3 – Other Dose Consequences	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE.	Conforms	With the exception that TID-14844 source term calculations will continue to be used in support of the current license basis for equipment qualification and radiation shielding systems.

	Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.		
4.4 – Acceptance Criteria	The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. The acceptance criteria for the various NUREG-0737 (Ref. 2) items generally reference General Design Criteria 19 (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).	Conforms	The EAB and LPZ acceptance criteria used are those of Table 6 of RG 1.183. The control room acceptance criterion is 5 rem TEDE. Updates to applicable criteria are included in this license amendment request to be consistent with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).
5. – ANALYSIS ASSUMPTIONS AND METHODOLOGY			
5.1 – General Considerations			
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Licensees should exercise caution in proposing deviations based upon data from a specific accident sequence since the DBAs were never intended to represent any specific accident sequence	Conforms	The analyses have been prepared, reviewed and will be maintained in accordance with quality assurance programs that comply with 10 CFR Part 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The dose analyses have not been limited to a specific set of accident progression assumptions.

	-- the proposed deviation may not be conservative for other accident sequences.		
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Conforms	Only safety-related mitigation features are credited in the dose analyses. Consideration of the loss of offsite power (LOOP) is taken in all accidents with regard to accident mitigation systems in order to maximize the release from a plant system. In general, the LOOP was used to limit equipment availability for plant cooldown, which in turn results in a larger amount of activity being released.
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be non conservative in another portion of the same analysis.	Conforms	Conservative parameters are assumed when calculating each contributor in the dose analyses.
5.1.4	In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calcuational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	Conforms	The analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.

5.2 – Accident-Specific Assumptions	Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST. The NRC staff has determined that the analysis assumptions in the appendices to this guide provide an integrated approach to performing the individual analyses and generally expects licensees to address each assumption or propose acceptable alternatives. The NRC will consider licensee proposals for changes in analysis assumptions based upon risk insights. The staff will not approve proposals that would reduce the defense in depth deemed necessary to provide adequate protection for public health and safety.	Conforms	The postulated accident radiological consequence analyses have been updated for AST. The DBA LOCA, FHA, MSLB, SGTR, LR, CRDE, and RVHD analyses have been analyzed. Assumptions have been addressed, as noted below. No changes have been made to analysis assumptions based upon risk insights.
5.3 – Meteorology Assumptions	Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. References 22 and 28 of this RG should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2 hour exposure period. The NRC computer code PAVAN implements Regulatory Guide 1.145 and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96 is generally acceptable to the NRC staff for use in determining control room χ/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident χ/Q values. Additional guidance is provided in Regulatory Guide 1.23.	Conforms	The atmospheric dispersion (χ/Q) values for the PBNP exclusion area boundary (EAB) and the low population zone (LPZ) are those from the current licensing basis. RG 1.194 has been used for onsite χ/Q values. Fumigation has not been included since no credit is taken for an elevated release. ARCON96 was used for determining onsite χ/Q values for onsite values. Meteorological data acquired in accordance with the PBNP meteorological measurement program for the five-year period from 2000 to 2005 is used to calculate onsite atmospheric dispersion. The onsite χ/Q methodology is based upon the methods in RG 1.194.

	All changes in χ/Q analysis methodology should be reviewed by the NRC staff.		
6. – ASSUMPTIONS FOR EVALUATING THE RADIATION DOSES FOR EQUIPMENT QUALIFICATION	The assumptions in Appendix I to this guide are acceptable to the NRC staff for performing radiological assessments associated with equipment qualification. The assumptions in Appendix I will supersede Regulatory Positions 2.c(1) and 2.c(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in Appendix I, all other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective.	Conforms	Generic Safety Issue (GSI) 187 has been resolved. The NRC staff concluded that there is no clear basis for a requirement to modify the design basis for equipment qualification to adopt AST since there would be no discernible risk reduction associated with such a requirement. Therefore, this license amendment request does not propose to modify the EQ design basis to adopt AST. The PBNP EQ analysis will continue to be based upon TID-14844 assumptions at this time.
Regulatory Guide 1.183 Appendix A: ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LWR LOSS-OF-COOLANT ACCIDENT			
SOURCE TERM ASSUMPTIONS			
1.	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1. The core inventory release fractions for the gap release and early in-vessel damage phases of the LOCA are consistent with Regulatory Position 3.2 and Table 2 of RG 1.183. See responses to Regulatory Position 3.
2.	If the sump or suppression pool Ph is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool Ph values less than 7 will be evaluated on a case-by-case basis. Evaluations of Ph should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The sump pH is controlled at a value greater than 7.0. The chemical form of the radioiodine released to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form.
ASSUMPTIONS ON TRANSPOPRT IN PRIMARY CONTAINMENT			
3.	Acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the		

	primary containment in PWRs or the drywell in BWRs are as follows:		
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange.	Conforms	The activity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment. The reactor coolant activity is assumed to be released over the first 30 seconds of the accident. The distribution does not need to be adjusted for internal compartment effects.
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. The prior practice of deterministically assuming that a 50% plate-out of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.	Conforms	For the containment leakage analysis, all activity released from the fuel is assumed to be in the containment atmosphere until removed by sprays, sedimentation, radioactive decay or leakage from the containment. Reduction of the airborne radioactivity in the containment by natural deposition is credited. The natural deposition removal coefficient for elemental iodine was determined to be 0.1/hr. Instantaneous plate-out is not assumed.
3.3	Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP may be credited. The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the un-sprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The	Conforms	Containment spray systems are designed and maintained in accordance with Chapter 6.5.2 of the SRP. Current FSAR radiological accident analyses do not take credit for operation of the containment spray system in the containment sump recirculation phase. The dose projections prepared in support of this submittal assume that containment spray is maintained throughout the injection phase of a LOCA, and continued during the early portions of the recirculation phase with no more than a 20 minute interruption. The ability to maintain spray during the early recirculation phase is essential as this is the period of highest iodine evolution from a damaged core. To support this assumption, it will be necessary for PBNP to alter the existing Emergency Operating Procedure(s) to direct continued containment spray while on sump recirculation if containment radiological conditions and/or core damage indicates it is required. While no interruption is anticipated, validation of the procedures will ensure that no more than a 20 minute interruption in

	SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays.		containment spray flow is incurred. The physical capability of the supporting systems will also need to be modified to ensure this capability is maintained. A loss of a single train of RHR would require supplying both core deluge and containment spray from the same RHR pump. The RHR and containment spray system will be modified. This will likely be in the form of modifying the containment spray pump discharge MOVs to permit throttling. The supporting analyses have de-rated the design spray flow rates to reflect a reduction in spray flow during the recirculation phase consistent with RHR pump NPSH limitations. The containment building is modeled as two discrete volumes: sprayed and unsprayed. The volumes are conservatively assumed to be mixed only by the containment fan coolers. 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 decontamination factor or DF of 200). With the RG 1.183 source term methodology this is considered as being 0.5 percent of the total inventory of elemental iodine that is released to the containment atmosphere over the duration of gap and in-vessel release phases. The particulate iodine removal rate is reduced by a factor of 10 when a DF of 50 is reached.
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	Conforms	PBNP does not have post-accident in-containment air filtration systems. PBNP does have containment air coolers which are credited for mixing.

3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid Ph is not maintained greater than 7.	N/A	This position relates to suppression pool scrubbing in BWRs. The position is not applicable to PBNP.
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).	N/A	This position relates to activity retention in ice condensers, which is not applicable to PBNP. The engineered safety features are addressed above.
3.7	The primary containment should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate.	Conforms	A containment leak rate, based on the proposed technical specifications, of 0.2% per day of the containment air is assumed for the first 24 hours. After 24 hours, the containment leak rate is reduced to 0.1% per day of the containment air.
3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	Conforms	The purge system is not considered to be in operation for this event.
ASSUMPTIONS ON DUAL CONTAINMENTS			
4.	For facilities with dual containment systems, the acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the secondary containment or enclosure buildings are as follows.	N/A	Regulatory Positions 4.1 through 4.6 apply to facilities with dual containment systems. As such, these positions are not applicable to PBNP.

ASSUMPTIONS ON ESF SYSTEM LEAKAGE			
5.	ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (Ref. A-7). The radiological consequences from the postulated leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of leakage from ESF components outside the primary containment for BWRs and PWRs.	Conforms	The radiological consequences from the postulated ESF systems leakage is analyzed and combined with consequences postulated for other fission product release paths.
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment should be assumed to instantaneously and homogeneously mix in the primary containment sump water at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used.	Conforms	Engineered Safety Feature (ESF) systems that recirculate water outside the primary containment (ECCS systems) are assumed to leak during their intended operation. Only iodine is released through this pathway since the noble gases are not assumed to dissolve in the sump and particulates would remain in the water of the ECCS leakage. It is assumed that the iodine is instantaneously and homogeneously mixed in the primary containment sump water at the time of release from the core.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump mini-flow return to the refueling water storage tank.	Conforms	Leakage from the ECCS system to the ESF rooms is taken as two times the FSAR allowable value of 400 cc/min. Recirculation is conservatively initiated at 0 minutes. The assumption of the ECCS leakage beginning at 0 minutes is not consistent with the assumption of injection spray termination in the containment leakage portion of the analysis. However, beginning the ECCS leakage at 0 minutes adds conservatism to the dose consequences. The leakage continues for the 30 day period following the accident considered in the analysis.

5.3	With the exception of iodine, all radioactive materials in the re-circulating liquid should be assumed to be retained in the liquid phase.	Conforms	With the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.
5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment.	Conforms	A flashing fraction of 7.75% was determined based on the temperature of the containment sump liquid at the time recirculation begins.
5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump Ph history and area ventilation rates.	Conforms	The iodine available for release at the time recirculation begins is based on the expected sump pH history and temperature. For the ECCS leakage to the auxiliary building, 10% of the total iodine in the leaked ECCS fluid is assumed to be available for release and is assumed to become airborne and leak directly to the environment from the initiation of recirculation through 30 days.
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable.	Conforms	For ECCS leakage into the auxiliary building, the form of the released iodine is 97% elemental and 3% organic. No credit for holdup, filtration or dilution of ECCS leakage into the auxiliary building is taken.

ASSUMPTIONS ON MAIN STEAM ISOLATION VALVE LEAKAGE IN BWRS			
6.	For BWRs, the main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The radiological consequences from postulated MSIV leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of MSIV leakage.	N/A	Regulatory Positions 6.1 through 6.5 relate to MSSV leakage in BWRs, which is not applicable to PBNP.
ASSUMPTION ON CONTAINMENT PURGING			
7.	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	Containment purge is not considered as a means of combustible gas or pressure control in this analysis. In addition, routine containment purge is not active for this event.

Regulatory Guide 1.183 Appendix B: ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT			
SOURCE TERMS			
1.	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms	An evaluation was performed to determine the impact on calculated dose results for the FHA where some nuclear fuel assemblies used in the PBNP FHA radiological analysis did not meet the Table 3 Footnote 11 criteria of RG 1.183. For this evaluation it was assumed that the fuel that did not meet the RG 1.183 Footnote 11 limits had higher gap fractions. The higher gap fractions were applied only to the fraction of the fuel that did not meet the limit, following a method previously approved by the NRC for Kewaunee Power Station (ML070430020). The gap fractions used were those from Safety Guide 25 with the value for I-131 adjusted consistent with the recommendation in NUREG/CR-5009. RG 1.183 Table 3 gap fractions used in the base analysis for fuel that met the Footnote 11 limits were then compared to the gap fractions that were applied for fuel that did not meet the Footnote 11 limits. The result of the calculations determined that in the event of a fuel handling accident, the dose limits for the EAB, LPZ and Control Room are met even if none of the fuel in the failed assembly met the RG 1.183 Footnote 11 limits. The dose results for both cases are provided in Section 6.6 of Enclosure 3.

No other changes to the NRC-approved FHA radiological analysis using AST for items described in RG 1.183 Appendix B.

Reference: USNRC Letter, "PBNP, Units 1 And 2 - Issuance Of Amendments Re: Technical Specification 3.9.3, Containment Penetrations, Associated With Handling Of Irradiated Fuel Assemblies and Use of Selective Implementation of the Alternative Source Term for Fuel Handling Accident," April 2, 2004. (ADAMS Accession ML040680918)

Regulatory Guide 1.183 Appendix E: ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR MAIN STEAM LINE BREAK ACCIDENT			
SOURCE TERMS			
1.	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.	Conforms	No fuel damage is postulated for the limiting event. See Item 2 below.
2.	If no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed.	Conforms	For the pre-accident iodine spike case it is assumed that a reactor transient has occurred prior to the MSLB and has raised the RCS iodine concentration to a conservative value of 60 μCi/gm of dose equivalent (DE) I-131. (The TS 3.4.16 limit for a transient is 50 μCi/gm of dose equivalent (DE) I-131.) For the accident-initiated iodine spike case, the primary system transient associated with the MSLB causes an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times the appearance rate corresponding to a maximum equilibrium RCS concentration of 0.5 μCi/gm of DE I-131 (proposed TS 3.4.16). The spike is allowed to continue until 4 hours from the start of the event. After this point in the accident there is no activity available for release from the gap. This is allowed per RG 1.183 guidance.
3.	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

4.	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms	Regulatory Position 4 - Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.
	TRANSPORT		
5.	Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows.		
5.1	For facilities that have not implemented alternative repair criteria (see Ref. E-1, DG-1074), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	The primary-to-secondary leak rate is 0.35 gpm per SG. PBNP has not implemented alternative repair criteria. The primary-to-secondary leak rate in the steam generators is assumed to be the leak rate Limiting Condition for Operation specified in the Technical Specifications. In addition, the leakage is apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS.
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	Based on the existing licensing basis, the primary-to-secondary leak rate is assumed to continue until the temperature of the leakage is less than 212°F at 30 hours. The release of radioactivity from the unaffected SG continues for 8 hours (time to place RHR in operation).

5.4	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.
5.5	The transport model described in this section should be utilized for iodine and particulate releases from the steam generators. This model is shown in Figure E-1 and summarized below.	Conforms	The transport model in this section is used for iodine and particulate release from the steam generators.
5.5.1	A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. During periods of steam generator dryout, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence.	Conforms	In the faulted SG, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. For the unaffected steam generator used for plant cooldown, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.
5.5.2	The leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" (Ref. E-2), during periods of total submergence of the tubes.	Conforms	Any postulated leakage that immediately flashes to vapor is assumed to rise through the bulk water of the SG into the steam space and is assumed to be immediately released to the environment.
5.5.3	The leakage that does not immediately flash is assumed to mix with the bulk water.	Conforms	All leakage that does not immediately flash is assumed to mix with the bulk water.
5.5.4	The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.	Conforms	The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. For the faulted SG, a partition factor of 1.0 is assumed for the iodine. For the intact SG, a partition factor of 0.01 is assumed for the iodine. The retention of particulate radionuclides in the unaffected SG is limited by the moisture carryover from the SG.

5.6	Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. E-3). The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.	Conforms	Steam generator tube bundle uncovery is not predicted or postulated for the intact SG.
Regulatory Guide 1.183 Appendix F: ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR STEAM GENERATOR TUBE RUPTURE ACCIDENT			
SOURCE TERM			
1.	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Conforms	No fuel damage is postulated to occur for the SGTR event. See response for Regulatory Position 3.
2.	If no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification. Two cases of iodine spiking should be assumed.	Conforms	No fuel damage is postulated to occur for the SGTR event. The two cases of iodine spiking in Regulatory Positions 2.1 and 2.2 are assumed.
2.1	A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).	Conforms	Case assumes a reactor transient prior to the postulated SGTR that raises the RCS iodine concentration to a conservative value of 60 μCi/gm of dose equivalent (DE) I-131. (The Technical Specification (TS 3.4.16) limit for a transient is 50 μCi/gm of dose equivalent (DE) I-131.) This is the pre-accident spike case.
2.2	The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be	Conforms	Case assumes the transient associated with the SGTR causes an iodine spike. The spiking model assumes the primary coolant activity is initially at the proposed Tech. Spec. value of 0.5 μCi/gm DE I-131. Iodine is assumed to be released from the fuel into the PCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. This is the accident-induced spike case.

	considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.		
3.	The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.
4.	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.
	TRANSPORT		
5.	Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows:		
5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	The primary-to-secondary leak rate is 0.35 gpm per SG. The primary-to-secondary leak rate in the steam generators is assumed to be the leak rate Limiting Condition for Operation specified in the Technical Specifications. In addition, the leakage is apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS.
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212° F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The release of radioactivity from the affected SG is assumed to continue for 30 minutes. The release of radioactivity from the unaffected SG is assumed to continue until shutdown cooling is in operation and the steam release from the SGs is terminated (8 hours into the event).
5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	The release of fission products from the secondary system is evaluated with the assumption of a coincident loss of offsite power (LOOP).
5.5	All noble gas radionuclides released from the primary system are assumed to be	Conforms	All noble gas activity carried over to the secondary side through SG tube

	released to the environment without reduction or mitigation.		leakage is assumed to be immediately released to the outside atmosphere.
5.6	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the steam generators is as follows: Appendix E, Regulatory Position 5.5.1 - All steam generators effectively maintain tube coverage. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing for all steam generators. Appendix E, Regulatory Position 5.5.2 - A portion of the primary-to-secondary ruptured tube flow through the SGTR is assumed to flash to vapor, based on the thermodynamic conditions in the reactor and secondary. The portion that flashes immediately to vapor is assumed to rise through the bulk water of the SG, enter the steam space, and be immediately released to the environment. Scrubbing of the flashed flow in the affected SG is credited. The methodologies presented in NUREG-0409 are used to determine the amount of scrubbing of the flashed flow. Appendix E, Regulatory Position 5.5.3 - All of the SG tube leakage and ruptured tube flow that does not flash is assumed to mix with the bulk water. Appendix E, Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG

			carryover rate of less than 1%. Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery is not postulated for this event for PBNP.
Regulatory Guide 1.183 Appendix G: ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR LOCKED ROTOR ACCIDENT			
SOURCE TERM			
1.	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	N/A	Analysis assumed all fuel rods in core are breached. The LR dose analysis was modeled using the gap fractions from Table 3 of RG 1.183 even though PBNP does not meet the limitations on maximum linear heat generation rate and burnup cited in Footnote 11 of RG 1.183. This condition was evaluated and it was determined acceptable due to the significant additional conservatisms applied in the LR radiological analysis.
2.	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment.	Conforms	See Regulatory Position 1 above.
3.	The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.
4.	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	It is assumed that 100 percent of the fuel rods in the core suffer damage as a result of the locked rotor sufficient that all of their gap activity is released to the reactor coolant system. Eight percent of the total I-131 core activity is in the fuel cladding gap. Ten percent of the total Kr-85 core activity is in the fuel cladding gap. Five percent of other iodine isotopes and other noble gases and 12 percent of the total core activity for alkali metals are assumed to be in the fuel cladding gap. Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.

5.	Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows:		
5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	The primary-to-secondary leak rate is 0.35 gpm per SG. The primary-to-secondary leak rate in the steam generators is assumed to be the leak rate Limiting Condition for Operation specified in the Technical Specifications. In addition, the leakage is apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS.
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212° F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The primary-to-secondary leakage is assumed to continue until shutdown cooling is in operation and the steam release from the SGs is terminated (8 hours into the event).
5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	Consideration of the loss of offsite power (LOOP) is taken in all accidents with regard to accident mitigation systems in order to maximizing the release from a plant system. In general, the LOOP was used to limit equipment availability for plant cooldown, which in turn, results in a larger amount of activity being released.
5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

5.6	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the steam generators is as follows: Appendix E, Regulatory Position 5.5.1 - All steam generators effectively maintain tube coverage. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing for all steam generators. Appendix E, Regulatory Position 5.5.2 - A portion of the primary-to-secondary ruptured tube flow through the SGTR is assumed to flash to vapor, based on the thermodynamic conditions in the reactor and secondary. The portion that flashes immediately to vapor is assumed to rise through the bulk water of the SG, enter the steam space, and be immediately released to the environment. Scrubbing of the flashed flow in the affected SG is credited. The methodologies presented in NUREG-0409 are used to determine the amount of scrubbing of the flashed flow. Appendix E, Regulatory Position 5.5.3 - All of the SG tube leakage and ruptured tube flow that does not flash is assumed to mix with the bulk water. Appendix E, Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.

			Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery is not postulated for this event for PBNP.
Regulatory Guide 1.183 Appendix H: ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR ROD EJECTION ACCIDENT			
SOURCE TERM			
1.	Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.	Conforms	The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3, and is provided in Table 5. In determining the doses following a rod ejection accident, it is assumed that 10% of the fuel rods in the core suffer sufficient damage as a result of departure from nuclear boiling (DNB) and all of their gap activity is released. It is assumed that 50 percent of the rods in DNB undergo centerline melting, with the melting limited to the inner 10 percent and occurring over 50 percent of the axial length, whereby 0.25 percent of the activity in the core is release as a result of partial melting of the fuel. Ten percent of the total core activity of iodine and noble gases and 12 percent of the total core activity for alkali metals are assumed to be in the fuel-cladding gap. The activity releases from the failed/melted fuel based on the Table 5 core average activities were multiplied by the maximum radial peaking factor of 1.8.
2.	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the loss-of-coolant accident (LOCA), main steam line break, and steam generator tube rupture.	Conforms	See Position 1 above.
3.	Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant	Conforms	For the containment release case, 100% of the activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the containment atmosphere. For the secondary release case, 100% of the activity released from the damaged fuel is assumed to mix

	and available for release to the secondary system.		instantaneously and homogeneously in the primary coolant and be available for leakage to the secondary side of the SGs.
4.	The chemical form of radioiodine released to the containment atmosphere should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. If containment sprays do not actuate or are terminated prior to accumulating sump water, or if the containment sump Ph is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case basis. Evaluations of Ph should consider the effect of acids created during the rod ejection accident event, e.g., pyrolysis and radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The chemical form of radioiodine released from the damaged fuel to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Containment sump pH is controlled at a value greater than 7.0.
5.	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	The chemical form of radioiodine released from the SGs to the environment is assumed to be 97% elemental iodine, and 3% organic iodide.
	TRANSPORT FROM CONTAINMENT		
6.	Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the containment are as follows.		
6.1	A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to Appendix A to this guide for guidance on acceptable methods and assumptions for evaluating these mechanisms.	Conforms	For the containment leakage case, sedimentation of alkali metal particulates in containment is credited. Containment spray is not credited.
6.2	The containment should be assumed to leak at the leak rate incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing. Leakage from sub atmospheric containments is assumed to be terminated when the containment is brought to a sub atmospheric condition as defined in technical specifications.	Conforms	The containment is assumed to leak at the proposed TS maximum allowable rate of 0.2% for the first 24 hours and 0.1% for the remainder of the event.

TRANSPORT FROM SECONDARY SYSTEM			
7.	Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the secondary system are as follows.		
7.1	A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The primary-to-secondary leak rate is 0.35 gpm per SG. The primary-to-secondary leak rate in the steam generators is assumed to be the leak rate Limiting Condition for Operation specified in the Technical Specifications. In addition, the leakage is apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized
7.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS.
7.3	All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.	Conforms	All of the noble gas released to the secondary side is assumed to be released directly to the environment without reduction or mitigation.
7.4	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	Compliance with Appendix E Sections 5.5 and 5.6 is discussed below: Appendix E, Regulatory Position 5.5.1 – Both steam generators are used for plant cooldown. Therefore, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing. Appendix E, Regulatory Position 5.5.2 - None of the SG tube leakage is assumed to flash for this event. Appendix E, Regulatory Position 5.5.3 - All of the SG tube leakage is assumed to mix with the bulk water. Appendix E, Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the

			partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%. Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery is not postulated for this event.
Regulatory Guide 1.183 Appendix I: ASSUMPTIONS FOR EVALUATING RADIATION DOSES FOR EQUIPMENT QUALIFICATION			
1. – 13.	This appendix addresses assumptions associated with equipment qualification that are acceptable to the NRC staff for performing radiological assessments. As stated in Regulatory Position 6 of this guide, this appendix supersedes Regulatory Positions 2.c.(1) and 2.c.(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (USNRC, June 1984), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in this appendix, other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective.	N/A	Regulatory Positions 1 through 13 apply to equipment qualification radiological analyses. Equipment qualification radiological analyses are not being submitted.

14.1.8 Loss of Reactor Coolant Flow

Flow Coastdown Events

A loss of coolant flow incident can result from a mechanical or electrical failure in one or more reactor coolant pumps, or from a fault in the power supply to these pumps. If the reactor is at power at the time of the incident, the immediate effect of loss of coolant flow is a rapid increase in coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly. The following trip circuits provide the necessary protection against a loss of coolant flow incident and are actuated by:

1. Low voltage on pump power supply bus;
2. Pump circuit breaker opening (low frequency on pump power supply bus opens pump circuit breaker); or
3. Low reactor coolant flow.

These trip circuits and their redundancy are further described in Section 7.2, Reactor Protection System.

Frequency decay for both reactor coolant pumps during full power operation is the most severe credible loss-of-coolant flow condition. For this condition reactor trip together with flow sustained by the inertia of the coolant and rotating pump parts will be sufficient to prevent fuel failure, reactor coolant system overpressure and prevent the DNB ratio from going below the limit value.

Method of Analysis

The following loss of flow cases are analyzed:

1. Loss of two pumps from a reactor coolant system, heat output of 1650.0 MWt with two loops operating; and
2. Loss of one pump from a reactor coolant system, heat output of 1650.0 MWt with two loops operating; and
3. Reactor coolant pump underfrequency event for both pumps with frequency decay rate of 3 Hz/sec, heat output of 1650.0 MWt with two loops operating. (Ref. 1)

The third case represents the worst credible coolant flow loss. The first two cases are less severe, with the second case being the least severe. Loss of one pump above 50% of full load is assumed to cause a reactor trip by a low flow signal. For the third case, flow decreases with the frequency to the low reactor coolant pump bus frequency setpoint (57.5 Hz) where the pumps will trip. Reactor trip for the third case is caused by a low flow signal.

The normal power supplies for the pumps are the two buses connected to the generator, each of which supplies power to one of the two pumps. When a generator trip occurs, the pumps are automatically transferred to a bus supplied from external power lines. Therefore, the simultaneous loss of power to all reactor coolant pumps is a highly unlikely event.

Following any turbine trip, where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately one minute. Since both pumps are not on the same bus, a single bus fault would not result in the loss of both pumps.

These transients are analyzed by three digital computer codes. First, the LOFTRAN code is used to calculate the loop and core flow during the transient, the time of the reactor trip based on the calculated flow, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code is used to calculate the minimum DNBR during the transient based on the heat flux from FACTRAN and the flow from LOFTRAN.

Initial Operating Conditions

Initial reactor power, RCS temperature and pressure are assumed to be at the most limiting nominal conditions, i.e. 100% power, maximum RCS temperature, and reduced pressure operation. Uncertainties in initial conditions are included in the DNBR limits as described in Reference 2.

Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used. The total integrated Doppler reactivity (power defect) between 0% and 100% power is assumed to be 0.016 Δk.

For cases 1 and 2 (loss of two or one reactor coolant pumps), the most-positive moderator temperature coefficient (MTC) limit for full-power operation (0 pcm/°F) is assumed, because this results in the maximum core power during the initial part of the transient, when the minimum DNBR is reached. This is made in conjunction with an end of cycle (EOC) DNB axial power shape assumed in THINC. The shape provides the most limiting minimum DNBR for the loss of flow events.

For case 3 (underfrequency), a split cycle approach is used with respect to the MTC and axial power shape. For both cycle portions, conservative MTCs and axial power shapes were assumed.

Flow Coastdown

The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance and the pump characteristics and is based on high estimates of system pressure losses.

No single active failure in the plant systems and equipment which are necessary to mitigate the effects of the accident will adversely affect the consequences of the accident during the transient. A conservatively evaluated overall heat transfer coefficient has been used in the analysis.

Fuel Type and SG Tube Plugging Level

The loss of coolant flow analysis is performed to bound operation with OFA or 422V+ fuel and an effective (i.e. sleeved and/or plugged) uniform steam generator tube plugging level of up to 10% for Units 1 and 2.

Results

Figure 14.1.8-1 shows the reactor coolant flow coastdown curves for a loss of both pumps. Figure 14.1.8-1 also shows the nuclear flux, RCS pressures, average channel heat flux, and hot channel heat flux transients for the coastdown of two pumps. The corresponding transients for the coastdown of one pump, and for the underfrequency event, are shown in Figures 14.1.8-2 and 14.1.8-3 respectively. Table 14.1.8-1 summarizes the sequence of events for these transients.

Conclusions

Since the minimum DNBR remains above the design DNBR limit for all cases, there is no cladding damage and no release of fission products into the reactor coolant. Therefore, once the fault is corrected, the plant can be returned to service in the normal manner. The absence of fuel failures would, of course, be verified by analysis of reactor coolant samples.

Locked Rotor Accident

A hypothetical transient analysis is performed for the postulated instantaneous seizure of a reactor coolant pump rotor. Flow through the reactor coolant system is rapidly reduced, leading to a reactor trip on a low-flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to heat up and expand. At the same time, heat transfer to the shell side of the steam generator is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the steam generator causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect is not included in the analysis.

Method of Analysis

Two digital computer codes are used to analyze this transient. The LOFTRAN code is used to calculate the resulting loop core and flow transients following the pump seizure, the time of reactor trip based on loop flow transients, nuclear power following reactor trip, and to determine peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN code, which uses the core flow and nuclear power calculated by LOFTRAN. The FACTRAN code includes a film boiling heat transfer coefficient.

One case is analyzed: one RCP coasting down, one locked rotor. At the beginning of the postulated locked rotor accident (i.e., at the time the shaft in one of the reactor coolant pumps is assumed to seize), the plant is assumed to be operating at 102 percent of NSSS Thermal Design Power, with maximum steady state pressure and maximum steady state coolant average temperature.

Then, peak pressure is evaluated; the initial pressure is conservatively estimated as 50 psi above nominal pressure (2250 psia) to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure. The pressure response shown in Figure 14.1.8-4 is the response at the point in the reactor coolant system having the maximum pressure.

Evaluation of the Pressure Transient - After pump seizure, the neutron flux is rapidly reduced by control rod insertion effect. Rod motion is assumed to begin one second after the flow in the affected loop reaches 87% of nominal flow. No credit is taken for the pressure-reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip. Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The lift pressure of the pressurizer safety valves is assumed to be 4% above the nominal set pressure of 2500 psia. Once this lift pressure is reached, an additional delay of 1 second is assumed to account for the clearing of the water in the pressurizer safety valve loop seals. The safety valve steam relief capacity is 288,000 lbm/hr per valve.

Evaluation of Departure from Nucleate Boiling in the Core During the Accident - For this accident, departure from the nucleate boiling is assumed to occur in the core, and therefore, an evaluation of the consequence with respect to fuel rod thermal transients is performed. Results obtained from analysis of this hot spot condition represent the upper limit with respect to cladding temperature and zirconium-water reaction. In the evaluation, the rod power at the hot spot is assumed to be 2.60 times the average rod power ($F_Q = 2.60$) at the initial core power level.

Film Boiling Coefficient - The film boiling coefficient is calculated in the FACTRAN code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature, which is the average between the wall and bulk temperatures. The program calculates the film coefficient at every time step, based on the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flow rate as a function of time are used as program input. For this analysis, the initial values of the pressure and the bulk density are used throughout the transient, since they are the most conservative with respect to cladding temperature response. For conservatism, departure from nucleate boiling is assumed to start at the beginning of the accident.

Fuel - Cladding Gap Coefficient - The magnitude and the time dependence of the heat transfer coefficient between fuel and cladding (gap coefficient) have a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between the pellet and the cladding. Based on investigations of the effect of the gap coefficient on the maximum cladding temperature during the transient, the gap coefficient is assumed to increase from a steady-state value consistent with an initial fuel temperature to 10,000 Btu per hour-square foot-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value is released to the cladding at the initiation of the transient.

Zirconium-Steam Reaction - The zirconium-steam reaction can become significant above a cladding temperature of 1800 °F. The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium-steam reaction:

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp \frac{(-45{,}500)}{1.986T}$$

where:

w	=	amount reacted (mg/cm^2)
t	=	time (seconds)
T	=	temperature (°K).

The reaction heat is 1510 cal/gm.

Results

Figure 14.1.8-4 shows the core flow and loop flow transients, the nuclear power and maximum pressure transients, the average channel and hot channel heat flux transients, and the cladding temperature transient. The results of these calculations are summarized in Table 14.1.8-2. The sequence of events is shown in Table 14.1.8-1.

Conclusions

Since the peak reactor coolant system pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted condition stress limits of 3120 psia, the integrity of the primary coolant system is not endangered.

Since the peak cladding surface temperature calculated for the hot spot during the more severe transient remains considerably less than 2700 °F and the amount of zirconium-water reaction is small, the core remains in place and intact with no consequential loss of core cooling capability.

Radiological Consequence of the 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 clad damage is assumed to occur as a result of the reduced flow. Due to the pressure differential between primary and secondary systems, and assumed steam generator tube leaks, 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 dump valves or main steam safety valves. In addition, it is postulated that some of the iodine activity contained in the secondary coolant prior to the accident is released to atmosphere as a result of steaming of the steam generators following the accident.

This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite and control room doses resulting from the release. The specific analyses conducted for the PBNP dose consequences were accepted by the NRC (Reference 4).

Input Parameters and Assumptions

The analysis of the locked rotor event radiological consequences used the analytical methods and assumptions outlined in the Standard Review Plan (References 5 and 7) and is documented in Reference 8.

The uprated power level of 1650 MWt is used in the analysis. For the pre-accident iodine spike, it is assumed that a reactor transient has occurred prior to the locked rotor and has raised the RCS iodine concentration 50 μCi/g of dose equivalent (DE) I-131. Since fuel failure is assumed for this accident, it is not necessary to also assume an accident initiated spike, as is the case for events without fuel failure such as a steam generator tube rupture or main steam line break. 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/g for gross radioactivity. The iodine activity concentration of the secondary coolant at the time the locked rotor occurs is assumed to be equivalent to the Technical Specification limit of 1.0 uCi/gm of DE-I131.

In determining the offsite and control room doses following the locked rotor, it is conservatively assumed that 100% of the fuel rods in the core suffer sufficient damage that all the their gap activity is released to the RCS. Ten percent of the total core activity for both iodines and noble gases is assumed to be in the fuel-cladding gap (Reference 6).

The total primary-to-secondary SG tube leak rate used in the analysis is 0.35 gpm (500 gallons/day) per steam generator or 0.70 gpm (1000 gallons/day) total. Technical Specifications provide a basis for this assumption by the establishment of: 1) a primary to secondary operational leakage limit of 150 gallons/day per Steam Generator, and 2) a Steam Generator Program which includes structural integrity and accident induced leakage performance criteria. (Reference 10) No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. An iodine partition factor in the SGs of 0.01 (Ci I/g steam)/(Ci I/g water) is used (Reference 7). All noble gas activity carried over the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

At eight hours 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 specific assumptions applied to PBNP are summarized in Table 14.1.8-3, 14.1.8-4 and 14.1.8-5. The thyroid dose conversion factors, breathing rates, and atmospheric dispersion factors used in the dose calculations are given in Table 14.1.8-3. The core and coolant activities used in the radiological calculations are given in Table 14.1.8-4. The remaining major assumptions and parameters used specifically in the locked rotor analysis are itemized in Table 14.1.8-5.

Control Room Model

For the locked rotor accident it is assumed that the HVAC system begins in Mode 1 (normal operating mode). The dose rates in the control room trip the control room monitors within 30 minutes, switching the system to Mode 4 (emergency mode) where it remains throughout the event. The control room doses are calculated over a period of 24 hours to ensure that the largest doses to the control room operator are calculated since the ventilation system will continue to operate in the specified mode for several hours following the termination of the steam releases. In addition, a factor of 10 reduction to the control room thyroid dose was applied due the assumption that the control room operator ingested potassium iodide pills. The parameters associated with the control room HVAC modes assumed for the locked rotor accident are summarized in Table 14.1.8-6. FSAR 9.8 provides a complete description of the control room HVAC system.

Acceptance Criteria

The dose limits for a locked rotor accident are a "small fraction" of 10 CFR 100 guideline values. A "small fraction" is considered 10% of 10 CFR 100 guideline values, or 30 rem thyroid and 2.5 rem whole body. The criteria defined in the Standard Review Plan Section 6.4 are used for control room dose limits: 30 rem thyroid, 5 rem whole body, and 30 rem beta skin. (Reference 9)

Results/Conclusions

The results of the offsite and control room dose analyses are provided in Table 14.1.8-2, and indicate that the acceptance criteria are met. The control room whole body dose, thyroid dose, and beta skin dose have been calculated to be within the acceptance criteria. Independent assessments (Reference 4) of the onsite and offsite radiological doses have concluded that the loss of coolant accident (LOCA) is more limiting than the RCP Locked Rotor event.

References

1. Goldberg, G., Westinghouse Electric Corporation, letter to E. J. Lipke, NPD, "Wisconsin Electric Power Company Point Beach Units 1 and 2 Final Reports for RCP Bus Frequency Decay Analysis," WEP-91-196, August 19, 1991.

2. Friedland, A. J., Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-Proprietary), April 1989.

3. WE Letter to NRC, NPL 97-0144, "Supplement to Technical Specifications Change Requests 188 and 189," dated April 2, 1997.

4. NRC Safety Evaluation Report (SER) dated July 1, 1997, "Issuance of Amendments for Technical Specification Change Requests 188 and 189."

5. NUREG-0800, Standard Review Plan 15.3.3 - 15.3.4, Revision 2, "Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break," July 1981.

6. US AEC Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

7. NUREG-0800, Standard Review Plan 15.6.3, "Radiological Consequences of a Steam Generator Tube Rupture (PWR)", Rev. 2, July 1981.

8. Westinghouse Calculation, CN-CRA-96-093, "Point Beach Locked Rotor Offsite and Control Room Doses for the Replacement Steam Generator Program and the Fuel Upgrade/Uprate Programs," Rev. 1.

9. NUREG-0800, Standard Review Plan 6.4, "Control Room Habitability System," Rev 2, July 1981.

10. Amendment Nos. 223 and 229 to Renewed Facility Operating License Operating License Nos. DPR-24 and DPR-27, respectively for Point Beach Nuclear Plant, Units 1 and 2, dated August 22, 2006.

TABLE 14.1.8-1
LOSS OF FORCED REACTOR COOLANT FLOW
TIME SEQUENCE OF EVENTS

Case	Event	Time (Seconds)
Complete Loss of Forced Reactor Coolant Flow	Both operating RCPs lose power and begin coasting down	0.0
	RCP undervoltage trip setpoint reached	0.0
	Rods begin to drop	1.5
Partial Loss of Reactor Coolant Flow (two loops operating, one RCP coasting down)	Coastdown begins	0.0
	Low Flow Reactor Trip	1.5
	Rods begin to drop	2.5
Underfrequency Event	Frequency decay begins and RCPs begin to decelerate	0.0
	Underfrequency setpoint reached; RCPs trip and begin coasting down	0.8
	Low RCS flow reactor trip setpoint reached	2.1
	Rods begin to drop	3.1
	Minimum DNBR occurs	4.6
Locked RCP Rotor	Rotor on one RCP locks	0.0
	Low RCS flow reactor trip setpoint reached	0.03
	Rods begin to drop	1.03
	Maximum RCS pressure occurs	3.4
	Maximum cladding temperature occurs	3.8

TABLE 14.1.8-2
SUMMARY OF LIMITING RESULTS FOR
LOCKED ROTOR ACCIDENT

Reactor Plant Results	
Maximum Reactor Coolant System Pressure	2873 psia
Maximum Cladding Temperature at Core Hot Spot	2124°F
Zr-H_2O Reaction at Core Hot Spot	1.1% by weight
Radiological Results	
Site Boundary (0 - 2 hr)	
Thyroid	15.6 Rem
Whole Body (gamma)	1.8 Rem
Low Population Zone (0 - 8 hr)	
Thyroid	10.0 Rem
Whole Body (gamma)	0.2 Rem
Control Room	
Thyroid	65.3 Rem [1]
Whole Body (gamma)	0.4 Rem
Beta Skin	11.0 Rem
Results are less limiting than LOCA	

[1] This calculated dose exceeds the 30 rem thyroid limit; however, assuming that the operators would be instructed to take the potassium iodide pills, this control room thyroid dose would be reduced to approximately 6.5 rem, which is within the limit.

TABLE 14.1.8-3
ASSUMPTIONS USED FOR DOSE ANALYSES
RCP LOCKED ROTOR ACCIDENT (14.1.8)
STEAM GENERATOR TUBE RUPTURE ACCIDENT (14.2.4)
MAIN STEAM LINE BREAK ACCIDENT (14.2.5)
CONTROL ROD EJECTION ACCIDENT (14.2.6)

DOSE CONVERSION FACTORS, BREATHING RATES, ATMOSPHERIC DISPERSION FACTORS

Isotope	Thyroid Dose Conversion Factors (rem/curie)
I-131	1.07 E6
I-132	6.29 E3
I-133	1.81 E5
I-134	1.07 E3
I-135	3.14 E4

Time Period	Breathing Rate (m^3 / second)
0 - 8 hr	3.47 E-4
8 - 24 hr	1.75 E-4
24 - 720 hr	2.32 E-4

Location	Atmospheric Dispersion Factors (second / m^3)
Site Boundary	
0 - 2 hr	5.0 E-4
Low Population Zone	
0 - 8 hr	3.0 E-5
8 - 24 hr	1.6 E-5
24 - 96 hr	4.2 E-6
96 -720 hr	8.6 E-7

TABLE 14.1.8-4
ASSUMPTIONS USED FOR DOSE ANALYSES
RCP LOCKED ROTOR ACCIDENT (14.1.8)
STEAM GENERATOR TUBE RUPTURE ACCIDENT (14.2.4)
MAIN STEAM LINE BREAK ACCIDENT (14.2.5)
CONTROL ROD EJECTION ACCIDENT (14.2.6)

CORE AND COOLANT ACTIVITIES

Nuclide	Total Core Activity at Shutdown (Ci)	Maximum Coolant Activity (based on 1% fuel defects) (μCi/gm)
I-131	4.4 E7	2.4 E0
I-132	6.3E7	2.4E0
I-133	9.0E7	3.8 E0
I-134	9.9E7	5.3 E-1
I-135	8.4 E7	1.9 E0
Kr-85	5.4 E5	6.9 E0
Kr-85m	1.2 E7	1.4 E0
Kr-87	2.3 E7	9.7 E-1
Kr-88	3.2 E7	2.7 E0
Xe-131m	4.7 E5	2.5 E0
Xe-133	8.9 E7	2.3 E2
Xe-133m	2.8 E6	4.2 E0
Xe-135	2.3 E7	7.4 E0
Xe-135m	1.7 E7	4.0 E-1
Xe-138	7.5 E7	5.9 E-1

TABLE 14.1.8-5
ASSUMPTIONS USED FOR DOSE ANALYSES
RCP LOCKED ROTOR ACCIDENT
CORE AND COOLANT ACTIVITIES

PARAMETER	VALUE
Initial Power	1650 MWt
RCS Noble Gas Activity Prior to Accident	1.0% Fuel Defect Level
RCS Iodine Activity Prior to Accident	50 μCi/gm of DE I-131
Radioactivity Released to RCS from Failed Fuel Activity (Noble Gas & Iodine)	100% of Core Gap
Fraction of Core Activity in Gap (Noble Gas & Iodine)	0.10
Secondary Coolant Activity Prior to Accident	1.0 μCi/gm of DE I-131
Total SG Tube Leak Rate During Accident	0.7 gpm
SG Iodine Partition Factor	0.01
Duration of Activity Release From Secondary System	8 hours
Offsite Power	Lost
Steam Release from SGs to Environment	206,000 lb (0 - 2 hr) 434,000 lb (2 - 8 hr)

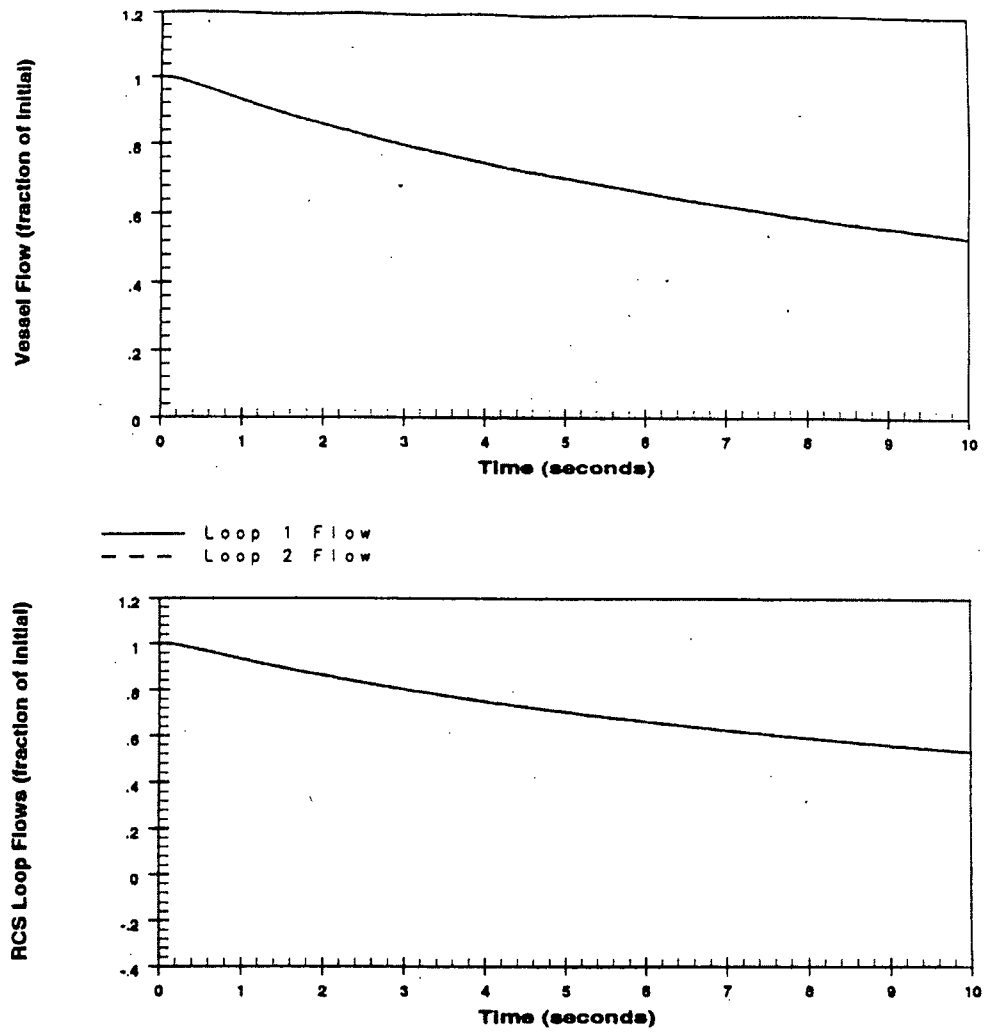
Reference

1. WE Letter to NRC, NPL 97-0144, "Supplement to Technical Specifications Change Requests 188 and 189", dated April 2, 1997.

TABLE 14.1.8-6
CONTROL ROOM PARAMETERS USED FOR DOSE ANALYSES
RCP LOCKED ROTOR ACCIDENT (14.1.8)
STEAM GENERATOR TUBE RUPTURE (14.2.4)
MAIN STEAM LINE BREAK ACCIDENT (14.2.5)
CONTROL ROD EJECTION ACCIDENT (14.2.6)

Volume	65,243 ft^3
CR HVAC Total Flow	19800 cfm
CR HVAC Mode 1	
Unfiltered Intake Flow	1000 cfm
Unfiltered Inleakage	65.2 cfm
CR HVAC Mode 4	
Filtered Intake Flow	4950 cfm
Unfiltered Intake Flow	10.0 cfm
Filter Efficiency	
Elemental	90%
Organic	90%
Particulate	99%
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4

FIGURE 14.1.8-1
Sheet 1 of 3
COMPLETE LOSS OF FLOW
(2/2 RCP COASTDOWN)

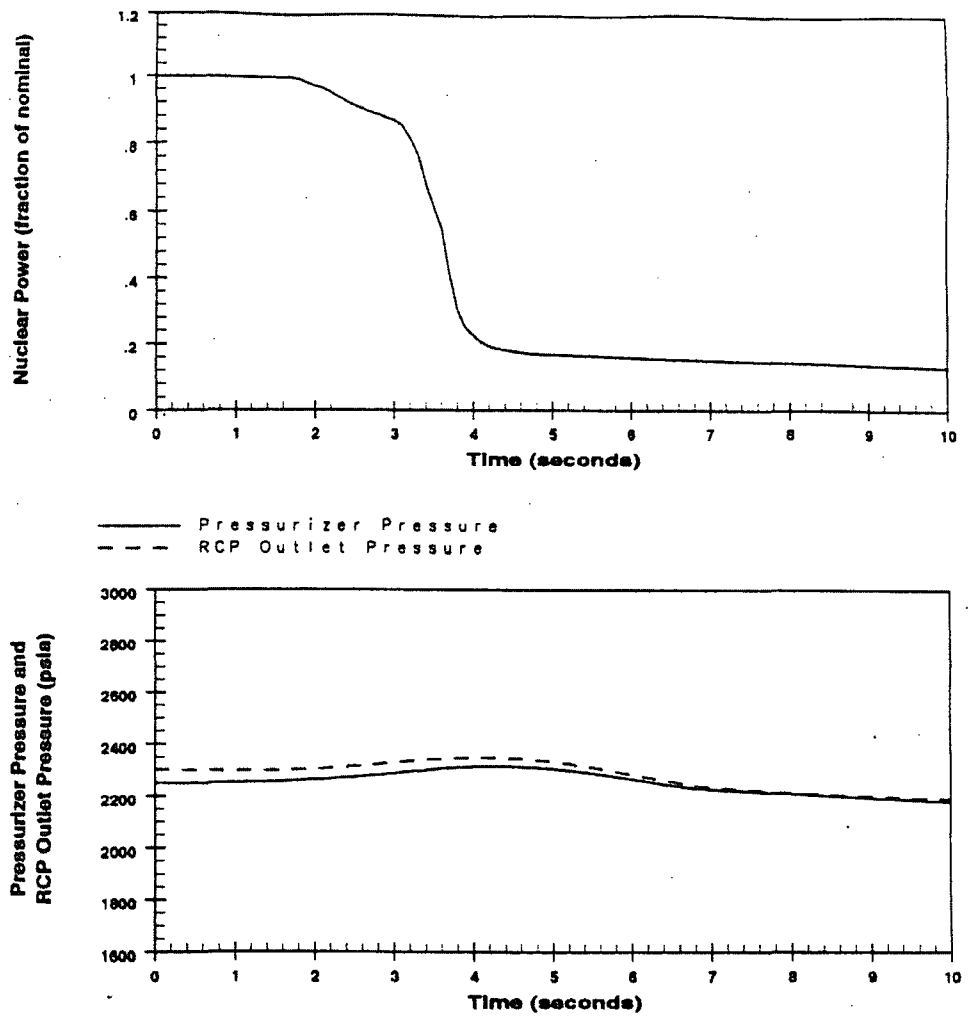


Point Beach Nuclear Plant
Units 1 and 2

Complete Loss of Flow
(2/2 RCP Coastdown)

Figure 14.1.8-1
(sheet 1 of 3)

FIGURE 14.1.8-1
Sheet 2 of 3
COMPLETE LOSS OF FLOW
(2/2 RCP COASTDOWN)

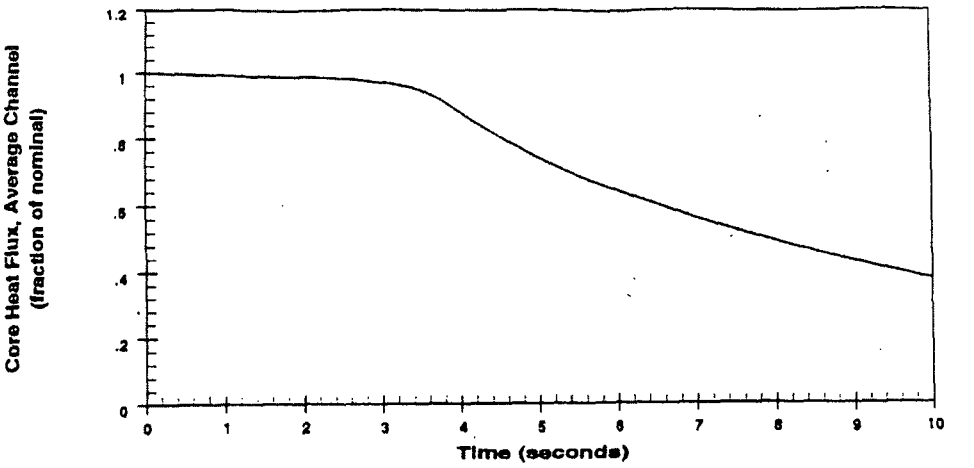


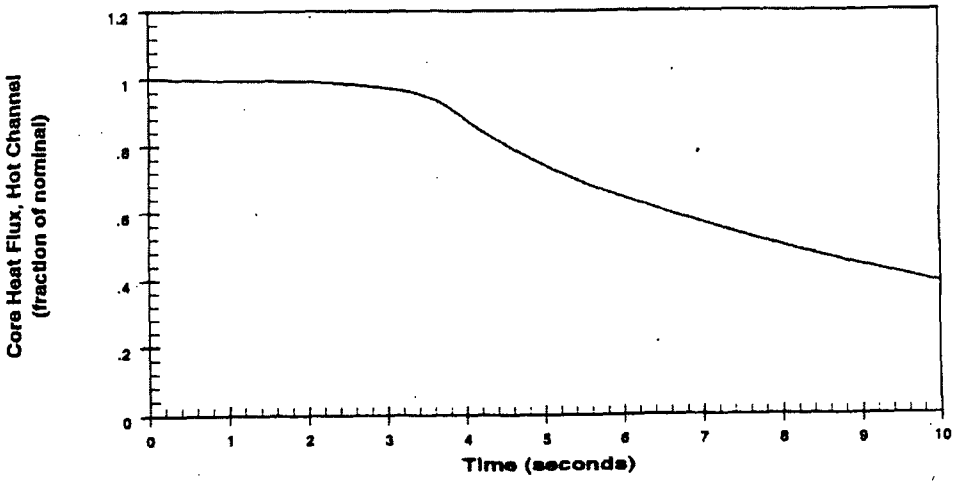
Point Beach Nuclear Plant
Units 1 and 2

Complete Loss of Flow
(2/2 RCP Coastdown)

Figure 14.1.8-1
(sheet 2 of 3)

FIGURE 14.1.8-1
Sheet 3 of 3
COMPLETE LOSS OF FLOW
(2/2 RCP COASTDOWN)



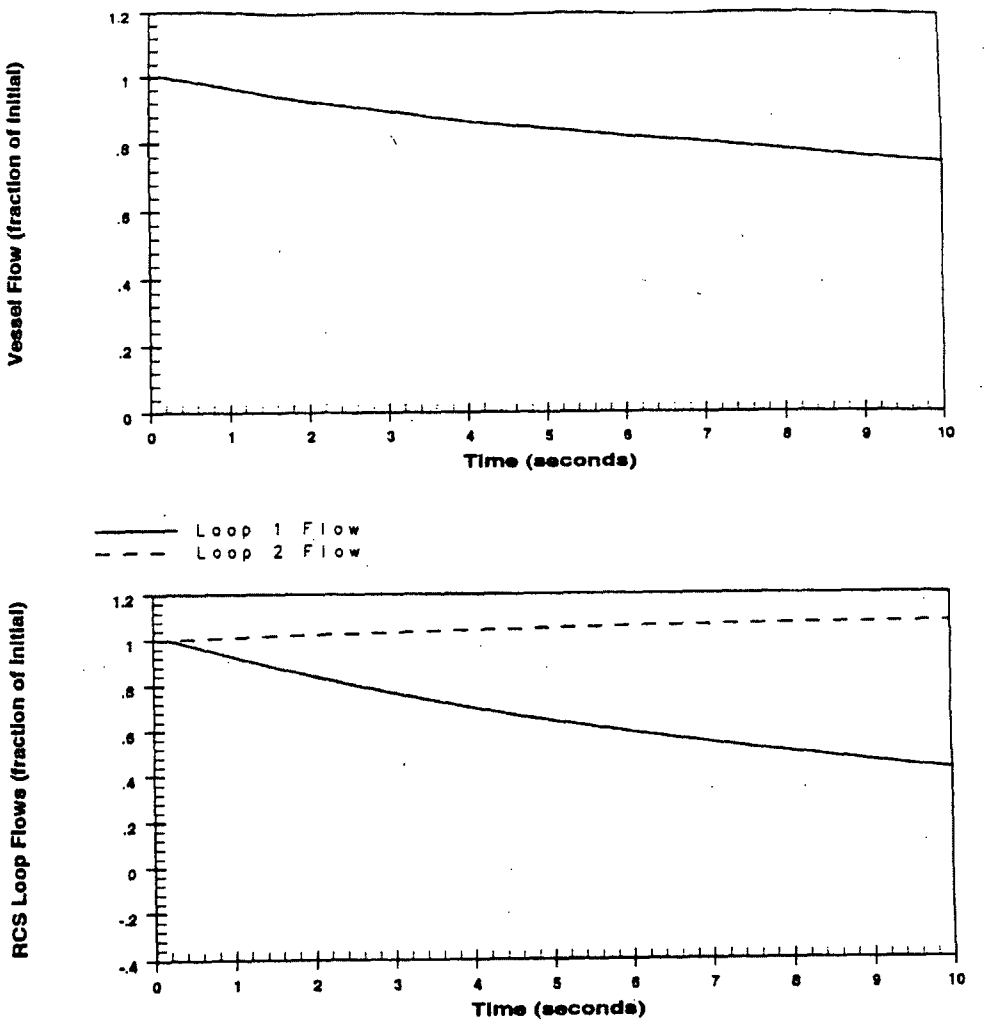


Point Beach Nuclear Plant
Units 1 and 2

Complete Loss of Flow
(2/2 RCP Coastdown)

Figure 14.1.8-1
(sheet 3 of 3)

FIGURE 14.1.8-2
Sheet 1 of 3
PARTIAL LOSS OF FLOW
(1/2 RCP COASTDOWN)

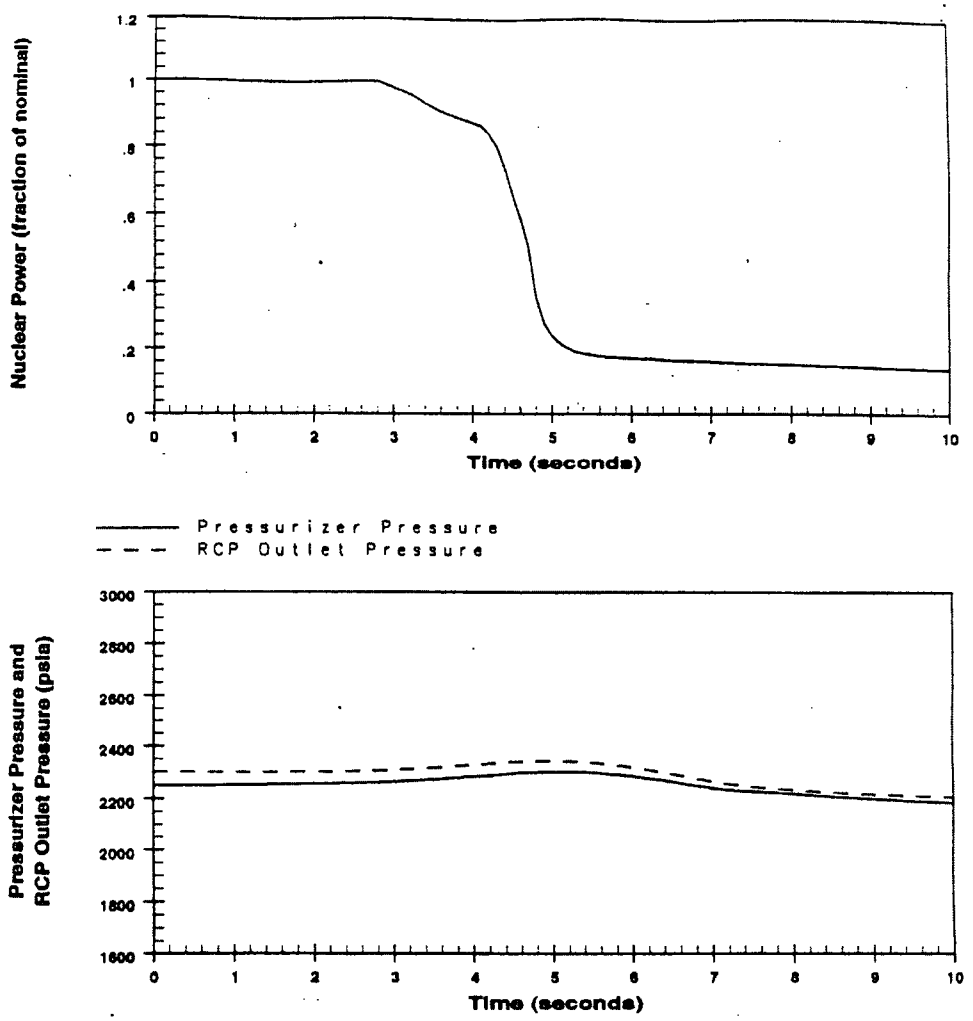


Point Beach Nuclear Plant
Units 1 and 2

Partial Loss of Flow
(1/2 RCP Coastdown)

Figure 14.1.8-2
(sheet 1 of 3)

FIGURE 14.1.8-2
Sheet 2 of 3
PARTIAL LOSS OF FLOW
(1/2 RCP COASTDOWN)

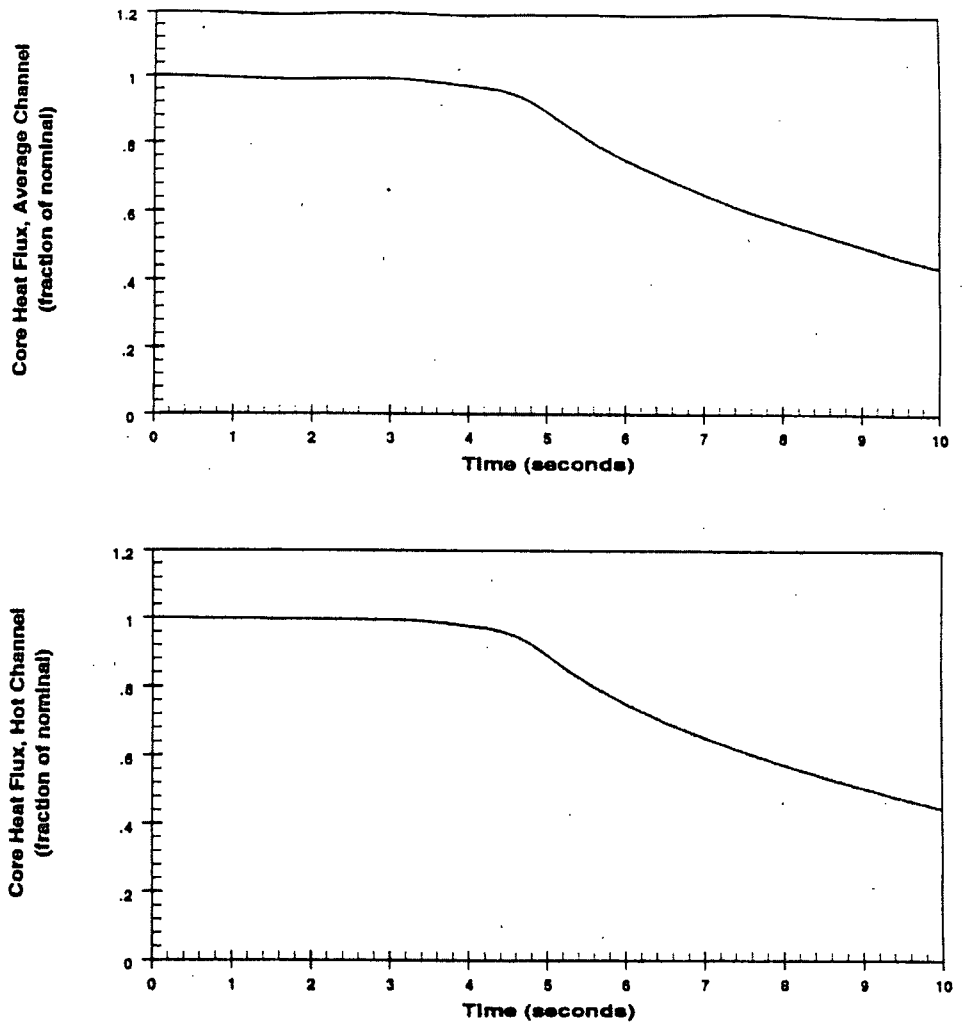


Point Beach Nuclear Plant
Units 1 and 2

Partial Loss of Flow
(1/2 RCP Coastdown)

Figure 14.1.8-2
(sheet 2 of 3)

FIGURE 14.1.8-2
Sheet 3 of 3
PARTIAL LOSS OF FLOW
(1/2 RCP COASTDOWN)

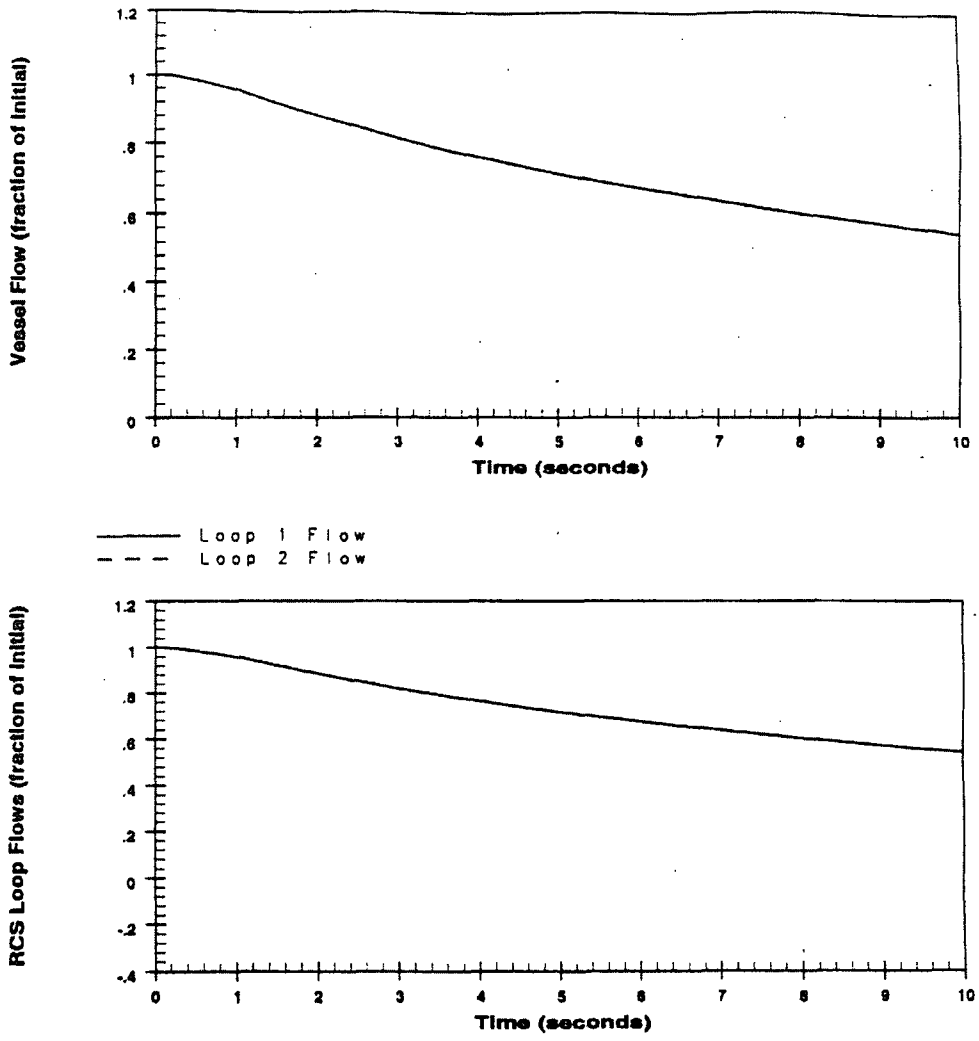


Point Beach Nuclear Plant
Units 1 and 2

Partial Loss of Flow
(1/2 RCP Coastdown)

Figure 14.1.8-2
(sheet 3 of 3)

FIGURE 14.1.8-3
Sheet 1 of 3
RCP BUS UNDERFREQUENCY
(2/2 RCP DECELERATION)

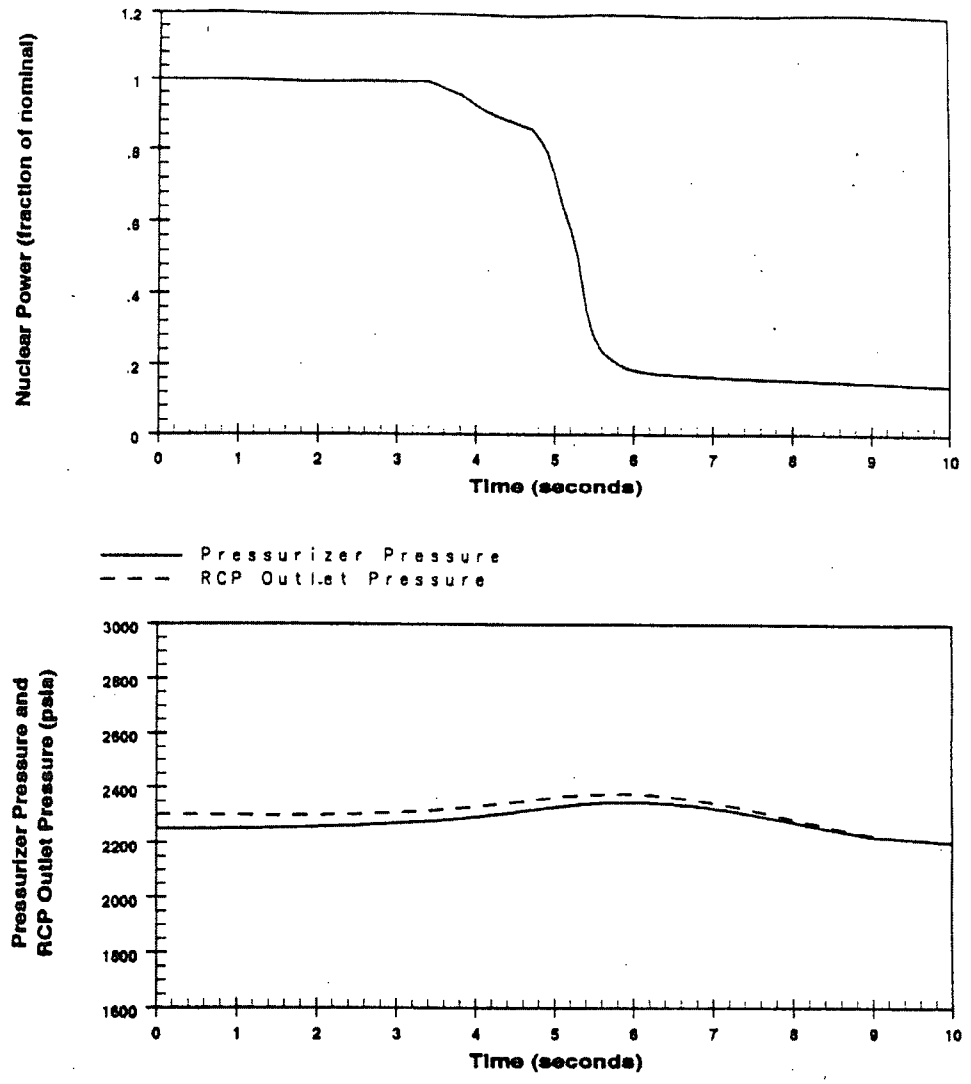


Point Beach Nuclear Plant
Units 1 and 2

RCP Bus Underfrequency
(2/2 RCP Deceleration)

Figure 14.1.8-3
(sheet 1 of 3)

FIGURE 14.1.8-3
Sheet 2 of 3
RCP BUS UNDERFREQUENCY
(2/2 RCP DECELERATION)

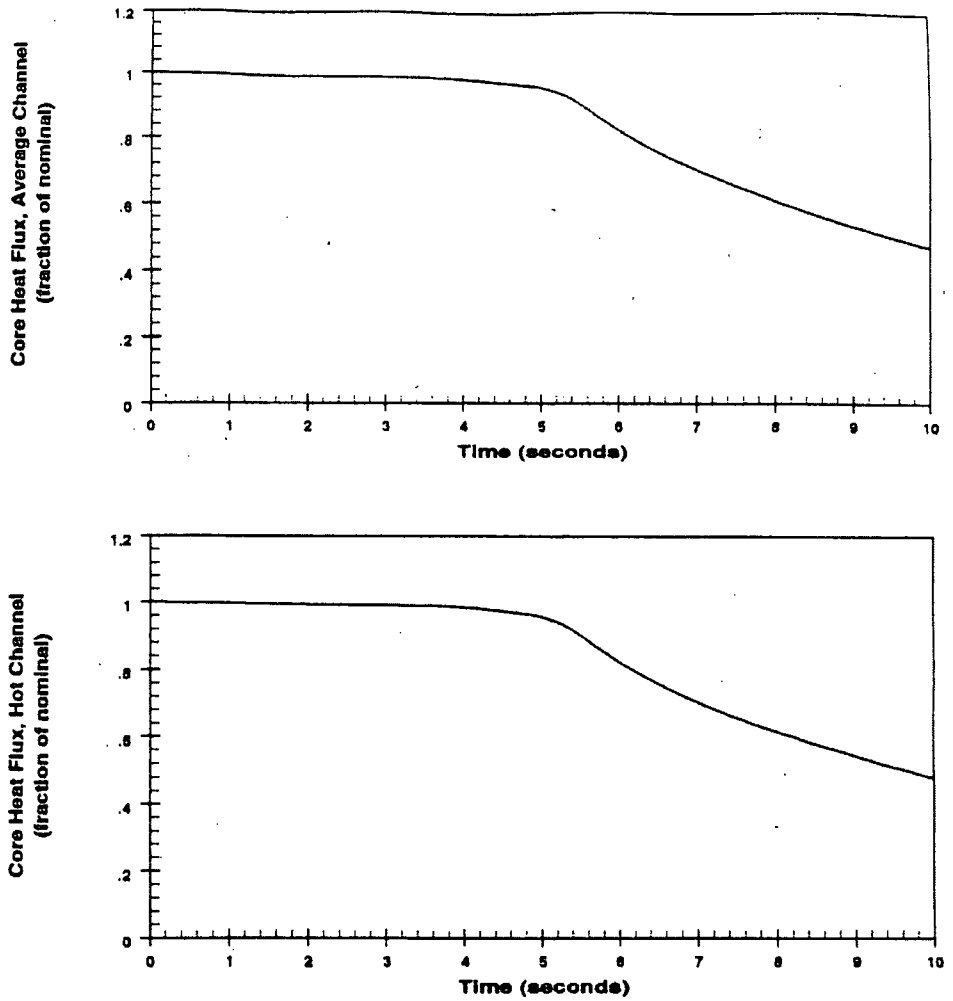


Point Beach Nuclear Plant
Units 1 and 2

RCP Bus Underfrequency
(2/2 RCP Deceleration)

Figure 14.1.8-3
(sheet 2 of 3)

FIGURE 14.1.8-3
Sheet 3 of 3
RCP BUS UNDERFREQUENCY
(2/2 RCP DECELERATION)

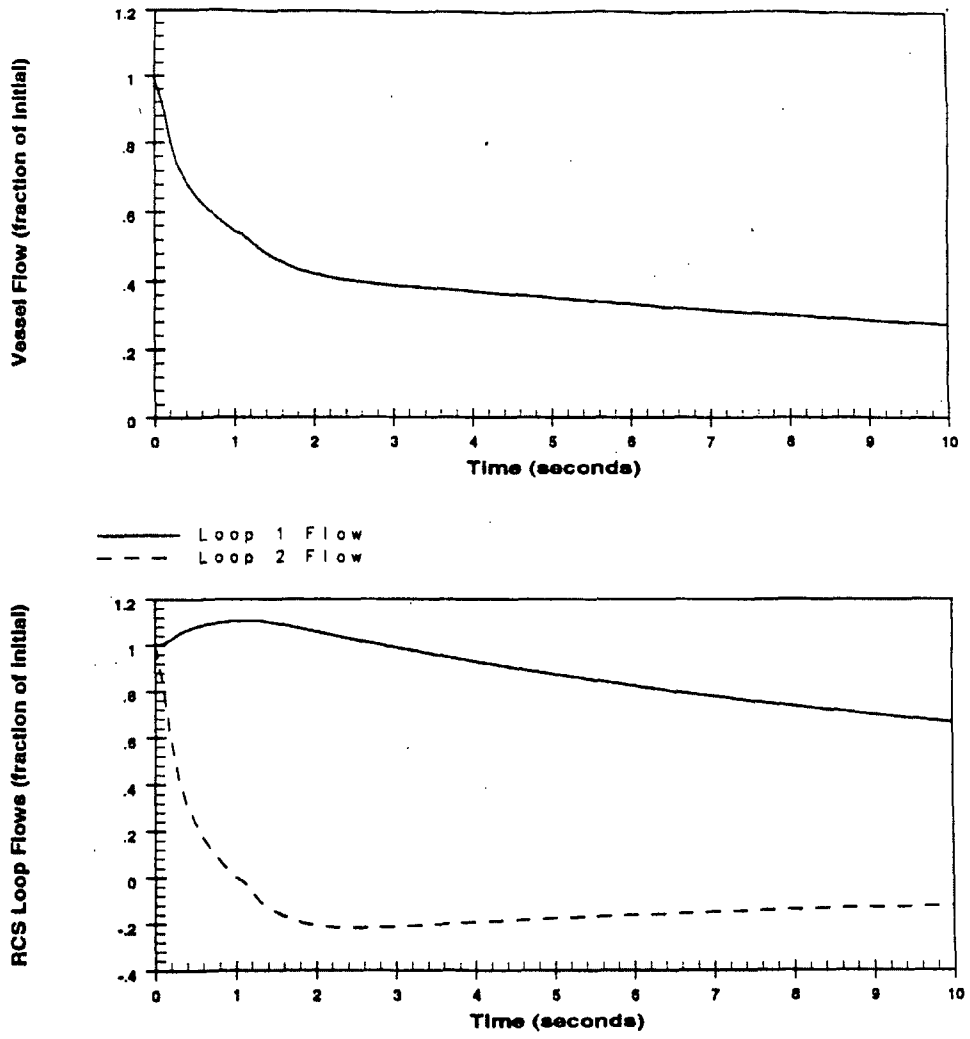


Point Beach Nuclear Plant
Units 1 and 2

RCP Bus Underfrequency
(2/2 RCP Deceleration)

Figure 14.1.8-3
(sheet 3 of 3)

FIGURE 14.1.8-4
Sheet 1 of 4
RCP LOCKED ROTOR

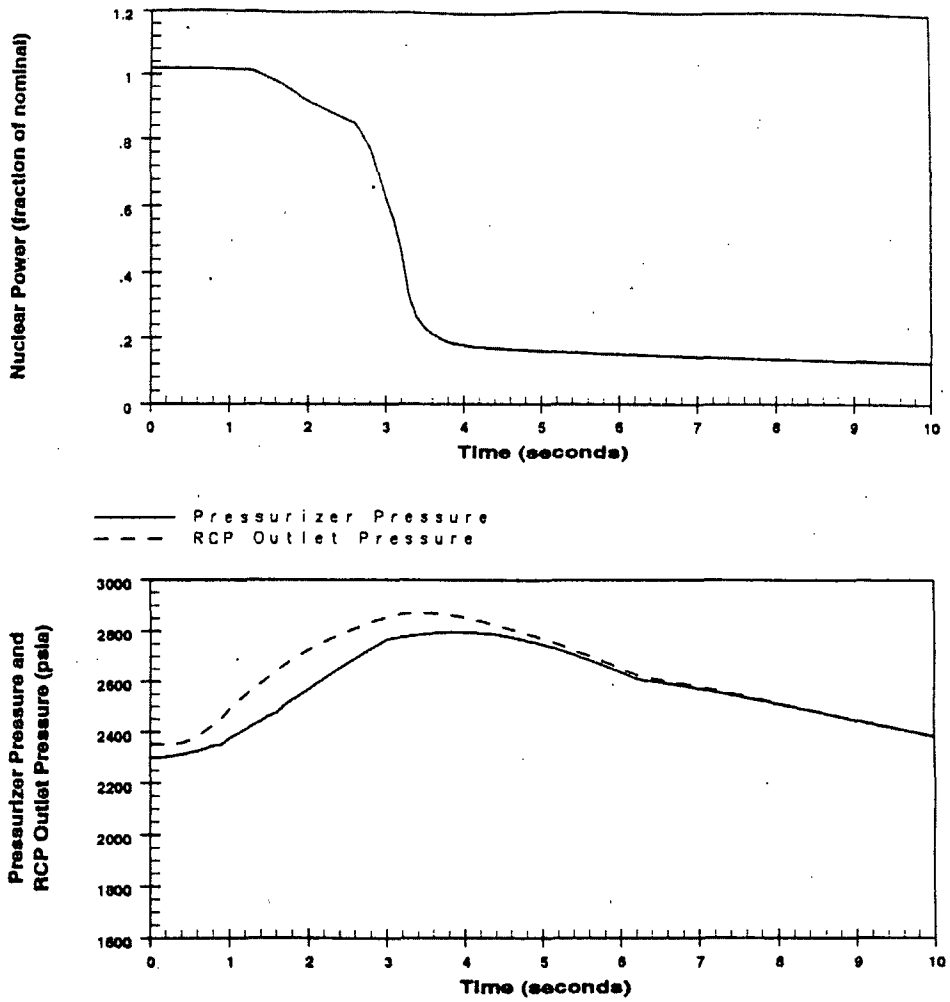


Point Beach Nuclear Plant
Units 1 and 2

RCP Locked Rotor

Figure 14.1.8-4
(sheet 1 of 4)

FIGURE 14.1.8-4
Sheet 2 of 4
RCP LOCKED ROTOR

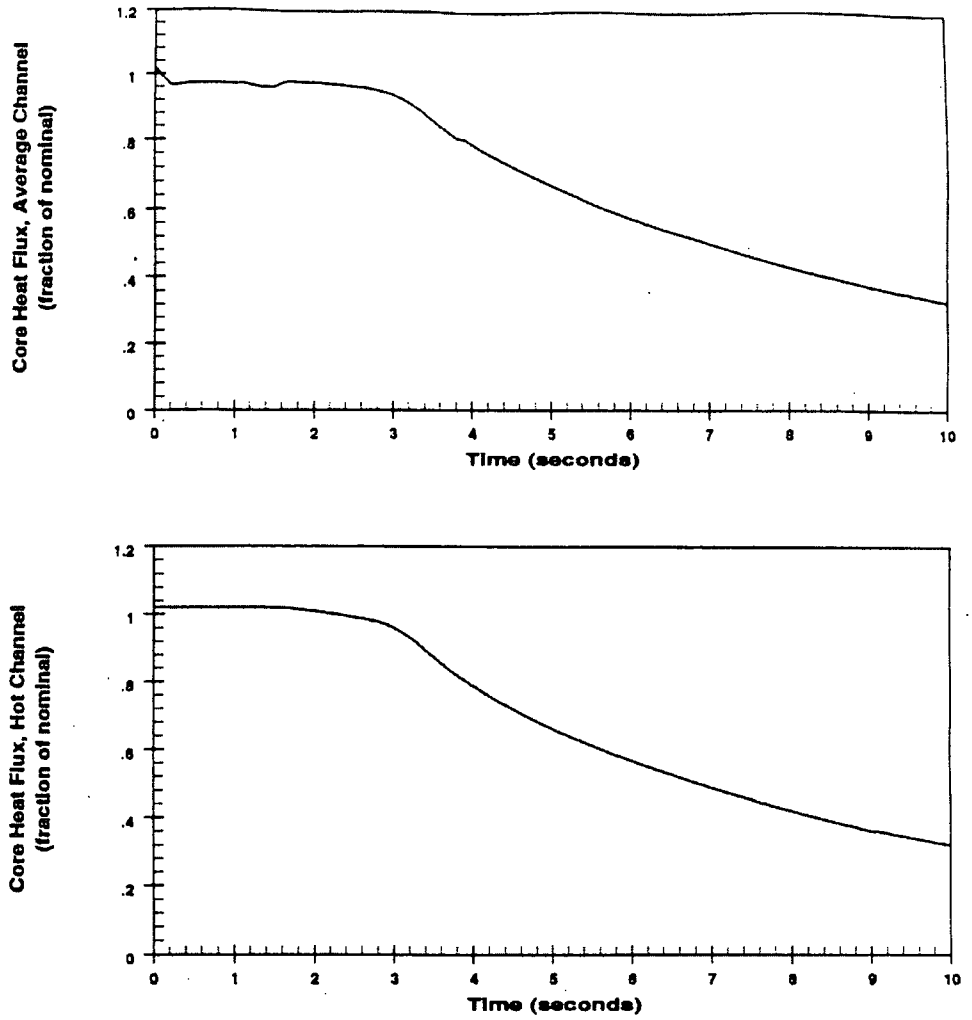


Point Beach Nuclear Plant Units 1 and 2

RCP Locked Rotor

Figure 14.1.8-4 (sheet 2 of 4)

FIGURE 14.1.8-4
Sheet 3 of 4
RCP LOCKED ROTOR

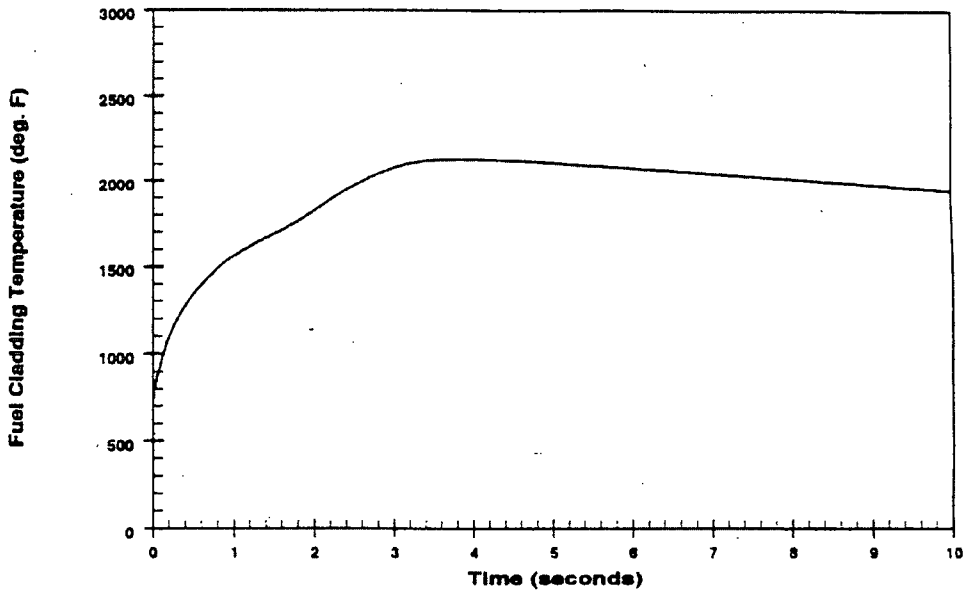


Point Beach Nuclear Plant Units 1 and 2

RCP Locked Rotor

Figure 14.1.8-4 (sheet 3 of 4)

FIGURE 14.1.8-4
Sheet 4 of 4
RCP LOCKED ROTOR



Point Beach Nuclear Plant
Units 1 and 2

RCP Locked Rotor

Figure 14.1.8-4
(sheet 4 of 4)

14.2 STANDBY SAFETY FEATURES ANALYSIS

Adequate provisions have been included in the design of the plant and its standby engineered safety features to limit potential exposure of the public to well below the limits of 10 CFR 100 for situations which have a very low probability of occurrence, but which could conceivably involve uncontrolled releases of radioactive materials to the environment. The situations which have been considered are:

1. Fuel Handling Accidents
2. Accidental Release of Waste Liquid
3. Accidental Release of Waste Gases
4. Rupture of a Steam Generator Tube
5. Rupture of a Steam Pipe
6. Rupture of a Control Rod Drive Mechanism Housing - Rod Cluster Control Assembly (RCCA) Ejection

14.2.1 Fuel Handling Accident

The following handling accidents are evaluated to ensure that no hazards are created:

1. A fuel assembly becomes stuck inside reactor vessel.
2. A fuel assembly or control rod cluster is dropped onto the floor of the reactor cavity or spent fuel pool.
3. A fuel assembly becomes stuck in the penetration valve.
4. A fuel assembly becomes stuck in the transfer carriage or the carriage becomes stuck.

The possibility of a fuel handling incident is very remote because of the many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety. Also, before any refueling operations begin, verification of complete rod cluster control assembly insertion is obtained by tripping the rods to obtain indication of rod drop and disengagement from the control rod drive mechanisms. Boron concentration in the coolant is raised to the refueling concentration and verified by sampling. Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical with all rod cluster assemblies withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications. As the vessel head is raised, a visual check is made to verify that the drive shafts are free in the mechanism housing.

After the vessel head is removed, the rod cluster control drive shafts are removed from their respective assemblies using the containment crane and the shaft unlatching tool. A load cell is used to indicate that the drive shaft is free of the control cluster as the lifting force is applied.

The fuel handling manipulators and hoists are designed so that fuel cannot be raised above a position which provides adequate shield water depth for the safety of operating personnel. This safety feature applies to handling facilities in both the containment and in the spent fuel pool area. In the spent fuel pool, the design of storage racks and manipulation facilities is such that:

> Fuel at rest is positioned by positive restraints in an eversafe, always subcritical, geometrical array, with no credit for boric acid in the water.
>
> Fuel can be manipulated only one assembly at a time.
>
> Violation of procedures by placing one fuel assembly in juxtaposition with any group of assemblies in racks will not result in criticality.

Adequate cooling of fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel assembly is immersed continuously while in the refueling cavity or spent fuel pool.

Even if a spent fuel assembly becomes stuck in the transfer tube, natural convection will maintain adequate cooling. The fuel handling equipment is described in detail in Section 9.

Two Nuclear Instrumentation System source range channels are continuously in operation and provide warning of any approach to criticality during refueling operations. This instrumentation provides a continuous audible signal in the containment, and would annunciate a local horn and a horn and light in the plant control room if the count rate increased above a preset low level.

Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical by at least 5% $\Delta\rho$ with all rod cluster control assemblies inserted. At this boron concentration, the core would also be more than 2% $\Delta\rho$ subcritical with all control rods withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications.

All these safety features make the probability of a fuel handling incident very low. Nevertheless, it is possible that a fuel assembly could be dropped during the handling operations. Therefore, this incident is analyzed both from the standpoint of radiation exposure and accidental criticality.

Special precautions are taken in all fuel handling operations to minimize the possibility of damage to fuel assemblies during transport to and from the spent fuel pool and during installation in the reactor. All handling operations on irradiated fuel are conducted under water. The handling tools used in the fuel handling operations are conservatively designed and the associated devices are of a fail-safe design.

In the fuel storage area, the fuel assemblies are spaced in a pattern which prevents any possibility of a criticality accident. In addition, the design is such that only one fuel assembly can be handled at a given time.

The motions of the cranes which move the fuel assemblies are limited to a relatively low maximum speed. Caution is exercised during fuel handling to prevent the fuel assembly from striking another fuel assembly or structures in the containment or fuel storage building. The fuel handling equipment suspends the fuel assembly in the vertical position during fuel movements, except when the fuel is moved through the transport tube.

The design of the fuel assembly is such that the fuel rods are restrained by grid clips which provide a total restraining force of approximately 60 lb. on each fuel rod. If the fuel rods are in contact with the bottom plate of the fuel assembly, any force transmitted to the fuel rods is limited due to the restraining force of the grid clips. The force transmitted to the fuel rods during fuel handling is not sufficient to breach the fuel rod cladding. If the fuel rods are not in contact with the bottom plate of the assembly, the rods would have to slide against the 60 lb. friction force. This would absorb the shock and thus limit the force on the individual fuel rods. After the reactor is shut down, the fuel rods contract during the subsequent cooldown and would not be in contact with the bottom plate of the assembly. Considerable deformation would have to occur before the rod would make contact with the top plate and apply any appreciable load on the fuel rod. Based on the above, it is felt that it is unlikely that any damage would occur to the individual fuel rods during handling. If one assembly is lowered on top of another, no damage to the fuel rods would occur that would breach the integrity of the cladding.

If during handling the fuel assembly strikes against a flat surface, the loads would be distributed across the fuel assemblies and grid clips and essentially no damage would be expected in any fuel rods. If the fuel assembly were to strike a sharp object, it is possible that the sharp object might damage the fuel rods with which it comes in contact, but breaching of the cladding is not expected.

The refueling operation experience that has been obtained with Westinghouse reactors has verified the expectation that no fuel cladding integrity failures occur during any fuel handling operations.

Rupture of all fuel elements in a withdrawn assembly is assumed as a conservative limit for evaluating the environmental consequences of a fuel handling incident. The remaining fuel assemblies are so protected by the storage rack structure that no lateral bending loads would be expected.

Radiological Consequences of a Fuel Handling Accident (FHA)

This section describes the assumptions and analyses performed to determine the potential offsite and control room radiological consequences for the postulated design basis fuel handling accident based on an Alternative Source Term (AST) in accordance with Regulatory Guide 1.183 (Reference 1). The analyses were performed such that the results are bounding for an accident occurring inside either containment or the spent fuel pool. (Reference 6 and 10) The NRC documented the acceptance of the analyses and results in Reference 2 based on the information provided in References 3, 4, and 5.

Input Parameters and Assumptions: The following assumptions were used in the analyses of the offsite and control room radiological consequences:

1. The reactor was assumed to have been operating at 1683 MWt prior to shutdown.

2. The reactor has been sub-critical for a minimum of 65 hours when the fuel handling accident occurs.

3. The fuel handling accident is assumed to result in damage to all of the fuel rods in the equivalent of one fuel assembly to the extent that all their gap activity is released.

4. The fission product gap inventories used are 8% for I-131, 10% for Kr-85, and 5% for all other noble gas and iodine nuclides. These values are based on Table 3 to Reference 1. These release fractions are acceptable for use because the PBNP core design is such that the peak burnup does not exceed 62,000 MWD/MTU and the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54 GWD/MTU.

5. The fission product inventory for the average fuel assembly at 65 hours after shutdown is provided in Table 14.2.1-1.

6. To account for differences in core power distribution across the core, the averaged fission product inventory in the dropped assembly is conservatively multiplied by a radial peaking factor of 1.8.

7. Consistent with the guidance of Reference 1, the iodine species in the pool is 99.85% elemental and 0.15% organic.

8. Consistent with the guidance of Reference 1, the effective decontamination factor (DF) used for iodine is 200 which accounts for scrubbing of the iodine as it evolves through the pool water. Applicability of this assumption is predicated on a minimum water level of 23 ft above the top of the reactor vessel flange and over the top of the assemblies in the spent fuel pool during movement of irradiated fuel assemblies. No DF is applied to the noble gas releases (i.e., no retention of the noble gases available for release) and an infinite DF is applied to the particulate radionuclides (i.e., the cesium and rubidium).

9. The activity released from the pool is assumed to be released from the containment refueling cavity or the spent fuel pool to the outside atmosphere over a two-hour period.

10. No credit is taken for removal of iodine by containment or spent fuel pool building ventilation systems' filters nor is credit taken for isolation of release paths. In addition, no credit is taken for the containment equipment hatch placement or closure nor is credit taken for having personnel air lock doors capable of closure.

11. The exclusion area boundary (EAB) and low population zone (LPZ) atmospheric dispersion factors values are found in Table 14.2.1-2. (Reference 12)

12. The control room atmospheric dispersion factor is based on a release from the Unit 2 containment building purge stack. This release point results in a bounding analysis because the assumptions and parameters used to model the activity released due to a FHA inside either containment are identical to those for a FHA in the spent fuel pool. The control room atmospheric dispersion factor was developed using ARCON96 (Reference 13) following the guidance provided in Draft Regulatory Guide DG-1111. (References 6 and 7) The control room atmospheric dispersion factor value is found in Table 14.2.1-2. The meteorological data set used to develop the control room atmospheric dispersion factor was collected at the site from 1997-1999. The use of this data set is acceptable only for use in the FHA control room dose assessment and is deemed not acceptable for use in amendments to this FHA dose assessment or other DBA dose assessments without further NRC staff review. (Reference 2)

13. Breathing rates assumed are consistent with Reference 1 and are listed in Table 14.2.1-2.

14. The total effective dose equivalent (TEDE) doses are determined at each location. The TEDE is equivalent to the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. Effective dose equivalent (EDE) is used in lieu of DDE in determining the contribution of external dose to the TEDE consistent with Reference 1. The dose conversion factors (DCFs) used in determining the CEDE dose are from the EPA Federal Guidance Report No. 11 (Reference 8). The dose conversion factors used in determining the EDE dose are from the EPA Federal Guidance Report No. 12 (Reference 9).

15. The site-boundary (also called the exclusion area boundary (EAB)) dose is calculated for the worst two-hour period and the low population zone (LPZ) dose is calculated for the release duration, that is two-hours for FHA (Reference 10). The control room personnel dose is calculated for 30 days assuming no administration of potassium iodide (KI). (Reference 6)

16. The control room HVAC system is assumed to be initially operating in normal mode, whereby fresh air is being brought into the control room unfiltered at a rate of 2000 cfm. It is conservatively assumed that the emergency HVAC mode is entered 10 minutes after event initiation based on the area monitor inside the control room reaching its alarm setpoint. The emergency HVAC mode is assumed to provide 4455 cfm of filtered outside air with no filtered recirculation.

17. Parameters used in the control room personnel dose calculations are provided in Table 14.2.1-2. These parameters include the normal operation flow rates, the post-accident operation flow rates, unfiltered inleakage rate, control room volume, filter efficiencies, and the control room operator breathing rates.

Acceptance Criteria

The EAB and LPZ dose Standard Review Plan (SRP) 15.0.1 (Reference 11) acceptance criteria for a fuel handling accident is 6.3 rem TEDE, which is approximately 25% of the 10 CFR 50.67 limit of 25 rem. The control room personnel dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67.

Results and Conclusions

The offsite and control room personnel doses due to a design basis FHA are presented below. (Reference 2, 6, 10) These doses are well within the acceptance criteria of SRP 15.0.1 and the dose limits of 10 CFR 50.67.

Location	Acceptance Criteria (rem)	TEDE (rem)
Exclusion Area Boundary	6.3	1.6
Low Population Zone	6.3	0.1
Control Room	5	2.8

14.2.1.1 Commitments

As documented in Reference 2 a commitment was made to follow the guidelines of NUMARC 93-01, Revision 3, Section 11.3.6, "Assessment Methods for Shutdown Conditions," Subsection 5, "Containment – Primary (PWR)/Secondary (BWR)," regarding decreasing doses further below that provided by natural decay.

In addition Reference 2 notes that the term "recently irradiated," as stated in Technical Specification 3.9.3, is a cycle specific number that represents the decay period for reduction in radionuclide inventory available for release in the event of an FHA. Therefore, as part of the cycle reload analysis a comparison of the cycle reload analysis assumptions should be made to those cited above to establish that the next cycle is bounded by the FHA analysis discussed in this section.

14.2.1.2 References:

1. USNRC, Regulatory Guide 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. USNRC Safety Evaluation Related to Amendment Nos. 213/218, "PBNP, Units 1 and 2 – Issuance of Amendments RE: Technical Specification 3.9.3, Containment Penetrations, Associated with Handling of Irradiated Fuel Assemblies and Use of Selective Implementation of the AST for FHA," April 2, 2004.

3. Letter, A.J. Cayia (NMC) to USNRC, "License Amendment Request 234, Selective Scope Implementation of Alternate Source Term for FHA," NRC 2003-0028, March 27, 2003.

4. Letter, A.J. Cayia (NMC) to USNRC, "Response to Request for Additional Information Regarding License Amendment Request 234, Selective Scope Implementation of Alternative Source Term for Fuel Handling Accident," NRC 2003-0106, October 30, 2003.

5. Letter, A.J. Cayia (NMC) to USNRC, "Supplemental Response to Request for Additional information Regarding License Amendment Request 234, Selective Scope Implementation of Alternative Source Term for Fuel Handling Accident," NRC 2003-0119, December 19, 2003.

6. PBNP Calculation 2003-0005, Revision 2, "Unit 2 Purge Stack Atmospheric Dispersion Factor Calculation and Control Room Dose Calculation for the Design Basis Fuel Handling Accident," December 17, 2003.

7. USNRC, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Draft Regulatory Guide DG-1111, December 2001.

8. USEPA, "Limiting values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation Submersion, and Ingestion," Federal Guidance Report No. 11, September 1988.

9. USEPA, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report No. 12, September 1993.

10. Westinghouse Calculation Note, "Point Beach - Fuel Handling Accident Doses Using Alternate Source Term - Control Room Sensitivities," CN-CRA-01-21, Revision 1, December 11, 2002.

11. Standard Review Plan (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternate Source Term," July 2000.

12. PBNP Calculation 96-0127, "PBNP Site Dispersion Factors for Accident Conditions," Revision 1, November 6, 1996.

13. J. V. Ramsdell, Jr. and C. A. Simonen, "Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Revision 1, May 1997.

TABLE 14.2.1-1
ACTIVITY IN AN AVERAGE FUEL ASSEMBLY AT 65 HOURS POST SHUTDOWN

Nuclide	Activity (Ci)
I-131	3.00E+05
I-132	3.05E+05
I-133	8.87E+04
I-135	7.81E+02
Kr-85m	4.31E+00
Kr-85	4.50E+03
Kr-88	3.45E-02
Xe-131m	3.93E+03
Xe-133m	1.45E+04
Xe-133	6.17E+05
Xe-135m	1.25E+02
Xe-135	1.26E+04

Note: Neither the gap fractions nor the radial peaking factor have been applied to these values.

TABLE 14.2.1-2
ASSUMPTIONS USED FOR THE FHA DOSE ANALYSIS

Parameter	Value
Core Power Level (1650 MWt x 1.02)	1683 MWt
Radial Peaking Factor	1.8
Number of Damaged Assemblies	1 assembly
Fission Product Decay Period	65 hr
Gap Fractions	
I-131	8% of activity
Kr-85	10% of activity
Other Iodine and Noble Gas	5% of activity
Water Level (minimum for reactor cavity or pool)	23 ft
Overall Pool Iodine Decontamination Factor	200
Noble Gas Decontamination Factor	1
Particulate Decontamination Factor	Infinite
Filter Efficiency	No filtration
Isolation of Release	No isolation
Atmospheric Dispersion Factors (X/Q)	
Exclusion Boundary Area	5.0E-04 sec/m^3
Low Population Zone	3.0E-05 sec/m^3
Control Room, Limiting Case – Unit 2 Purge Stack	5.76E-03 sec/m^3
Breathing Rate	3.5E-04 m^3/sec
Control Room HVAC Parameters	
Normal Mode Ventilation Flow Rates	
Filtered Makeup Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Makeup Flow Rate	2000 cfm
Unfiltered Inleakage Flow Rate	500 cfm
Emergency Mode Ventilation Flow Rates	
Filtered Makeup Flow Rate	4455 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Makeup Flow Rate	0 cfm
Unfiltered Inleakage Flow Rate	500 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%
Control Room Isolation Actuation Signal/Timing	
Area Monitor High Set-point	2 mrem/hr
Timing of High Radiation Signal	10 min
Occupancy Factors	
0-24 hours	1.0
1 – 4 days	0.6
4 – 30 days	0.4

14.2.4 Steam Generator Tube Rupture

General

A complete single tube break adjacent to the tube sheet in a steam generator is examined for two assumed situations. Since the reactor coolant pressure is greater than the steam generator shell side pressure, the contaminated reactor coolant discharges into the secondary system.

The activity release is limited by operator action to limit the primary to secondary fluid leakage and terminate the releases from the affected steam generator to the atmosphere. The analysis of the offsite consequences assumes that the ruptured steam generator is isolated thirty minutes after the postulated tube rupture accident.

Method of Analysis

A detailed time sequence of events is presented from occurrence of the assumed steam generator tube rupture until the primary to secondary break flow and release from the affected steam generator to the atmosphere have been terminated. Resultant radionuclide releases to atmosphere have been evaluated assuming that off-site power is lost (i.e., no credit for the condenser to mitigate the release).

The potential for an increased radioactive release to the environment due to steam generator tube bundle uncovery has been evaluated (Ref. 3). Uncovery does not significantly increase the radiological consequences associated with the SGTR accident. The probability of a significant release due to non-SGTR events, including the effects of tube uncovery, is sufficiently low to exclude such events from consideration. The NRC agrees with the position that the effects of partial steam generator tube bundle uncovery on the iodine release for SGTR and non-SGTR events is negligible (Ref. 4).

Analysis Assuming Minimum Auxiliary Feedwater and Off-Site Power are Available

The sequence of events following tube rupture is as follows:

1. Primary leakage takes place initially at a high rate (~87 lbm/sec) but rapidly drops to a lower leakage rate (~55 lbm/sec).

2. Within seconds, air ejector discharge of radionuclides may be alarmed by the auxiliary building vent stack monitor, thus warning the operator.

3. Pressurizer water level will continue to decrease for up to approximately five minutes before an automatic low pressure trip will occur. Normal core protection trips are available to prevent core damage during this time and may cause an earlier trip.

4. Safety injection is automatically actuated on low pressurizer pressure and the reactor coolant system is borated by the high head safety injection pumps.

5. Isolation of the affected steam generator can be achieved within 30 minutes by:

   a. Identifying the affected steam generator by early observation of rising liquid level, observing steam line radiation monitors, or analysis of a steam generator liquid sample.
   b. Closing the steam line isolation valve connected to the affected steam generator.
   c. Securing the auxiliary feedwater flow to that steam generator.

6. Other procedures to terminate primary-to-secondary leak flow through the ruptured tube include:

   a. Operator-controlled steam dumping to the condenser in order to; (1) reduce the reactor coolant temperature; (2) maintain primary coolant subcooling; (3) to minimize steam discharge from the affected steam generator.
   b. Operator-controlled RCS depressurization to restore reactor coolant inventory.
   c. Termination of safety injection in order to equalize pressure between the primary system and the affected steam generator.

7. Should the affected steam generator main steam isolation valve not close, the main steamline dump valves would be closed and atmospheric relief from the unaffected steam generator would be used for steam dumping (plant cooldown). (This case is described below.)

8. The unit has been cooled down and depressurized to an equilibrium condition where no further reactor coolant is discharged to the affected steam generator or steam released from the affected steam generator to the atmosphere is terminated.

Analysis Assuming Availability of Minimum Auxiliary Feedwater Without Off-Site Power

For the purpose of this analysis, it is assumed that when reactor trip occurs station normal power is lost. The reactor coolant pumps will then coast down and the condenser circulating water pumps will stop. On-site emergency power is available from the diesel generators to supply the necessary engineered safeguards equipment.

Core decay heat is then removed by natural circulation of reactor coolant to the steam generators. The atmospheric steam relief valves will open automatically to relieve high pressure in the steam generators. Steam dump to the condenser is isolated when condenser vacuum is lost. During this time, secondary safety valves may also lift.

Main steam safety valves open to restore primary system temperature to the hot shutdown value. They are designed to blowdown to 12.6% below the setpoint pressure to remove decay heat while maintaining the hot shutdown (hot standby per Technical Specification definitions) system pressure (Reference 9). With no operator action, the main steam safety valves would maintain the primary system temperature between approximately 540 and 557°F.

The safety injection system borates the reactor coolant system within several minutes and will eventually refill the reactor coolant system and pressurize it to a pressure at which the injection flow is balanced by discharge through the broken tube. Initially, the water level in the unaffected steam generator will decrease because the auxiliary feedwater supply will not match the steam relief needed to reduce the reactor coolant system to no-load temperature. When the steam dump is reduced to balance decay heat, the auxiliary feedwater supply exceeds decay heat requirements and the liquid level in the unaffected steam generator will increase. Because of the discharge from the reactor coolant system, the rate of increase in liquid level is greatest in the ruptured steam generator.

Up to this point, automatic actions will ensure safe shutdown of the reactor. Automatic actuation of safety injection will ensure that the core will not be damaged, and thus limit radioactivity releases to the level of the concentrations in the reactor coolant.

Within 30 minutes, the safety injection system will have refilled the reactor coolant system and the operator will have time to determine which steam generator is ruptured by observing steam generator levels, steam line radiation monitor, or analysis of a steam generator liquid sample. After the initial transient, the operator would isolate the affected steam generator, and perform a limited cooldown to assure subcooling margin. The safety injection system will maintain reactor coolant system pressure and pressurizer level, compensating for losses due to discharge in reaching pressure equilibrium between the reactor coolant system and the now isolated faulty steam generator and for contraction losses during the remainder of cooldown. After cooldown, RCS depressurization would be performed to restore reactor coolant inventory, and subsequently the safety injection flow would be terminated to stop the primary-to-secondary break flow.

After the primary-to-secondary break flow has been terminated, the RCS would be cooled down to cold shutdown conditions. The cooldown is initiated by manually controlling the steam relief on the unaffected steam generator. The relief valve and auxiliary feedwater pump capacities are adequate to cool down at 50°F/hour. At this rate, approximately 4 hours are required to cool down and depressurize the system to 350 psia, at which time the residual heat removal loop can be used to complete cooldown. During the cooldown, no further activity is discharged from the isolated steam generator.

Radiological Consequences of a Steam Generator Tube Rupture Accident

This section presents an evaluation of the offsite consequences of a steam generator tube rupture accident. The specific analyses conducted for the PBNP SGTR offsite consequences were accepted by NRC (Reference 7) and later revised by Reference 10, which corrected non-conservative input parameters used to calculate the accident-initiated iodine spike. The analyses of control room habitability determined that the loss of coolant accident was the most limiting event for control room habitability (FSAR 14.3.5, Reference 7 and Reference 11). Therefore, a specific description of the control room habitability assessment for a SGTR is not provided herein; however, the general results are provided.

Assumptions: The following assumptions were used in the analysis of the offsite consequences:

1. Both pre-accident and accident initiated iodine spikes are analyzed. For the pre-accident iodine spike, it is assumed that a reactor transient has occurred prior to the steam generator tube rupture and has raised the RCS iodine concentration to 50 μCi/gm of dose equivalent (DE) I-131. For the accident initiated iodine spike, the reactor trip associated with the steam generator tube rupture 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 0.8 μCi/gm of DE I-131. The duration of the accident initiated iodine spike is 1.6 hours. (Reference 5)

2. The noble gas activity concentration in the RCS at the time the accident occurs is based on a fuel defect of 1.0%. This is approximately equal to the Technical Specification value of 100/E-bar μCi/gm for gross radioactivity.

3. The iodine activity concentration of the secondary coolant at the time the steam generator tube rupture occurs is assumed to be equal to the Technical Specification limit of 1.0 μCi/gm of I-131.

4. The amount of primary to secondary steam generator tube leakage in the intact steam generator is assumed to be equal to 500 gallons/day (0.35 gpm). Technical Specifications provide a basis for this assumption by the establishment of: 1) a primary to secondary operational leakage limit of 150 gallons/day per Steam Generator, and 2) a Steam Generator Program which includes structural integrity and accident induced leakage performance criteria. (Reference 12)

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

6. An iodine partition factor in the steam generators is used as follows: 0.01 (curies Iodine / gm steam ÷ curies Iodine / gm water)

7. All noble gas activity carried over to the secondary side is assumed to be immediately released to the outside atmosphere.

8. Thirty minutes after the postulated tube rupture accident, the steam release from the ruptured steam generator is terminated. Approximately 123,600 lbs. of reactor coolant is discharged to the secondary side of the ruptured steam generator in 30 minutes. Also, approximately 74,000 lbs. of steam is released to the atmosphere via the ruptured steam generator during the time interval. The analysis also assumes that primary-to-secondary break flow is terminated in thirty minutes. However this assumption does not impact the offsite (or the control room) radiological consequences because the release from the ruptured generator is terminated in 30 minutes. Continued break flow past this time does not contribute to the release.

9. Auxiliary feedwater is available during the accident.

10. Pressure between the ruptured steam generator and the primary system is equalized such that the ruptured steam generator is not overfilled.

11. Eight hours after the accident the residual heat removal system is assumed to be placed in service and there are no further steam releases to the atmosphere from the secondary system. (Note that this is an analysis assumption in the approved radiological analysis that terminates steam releases. The RHR safety-related and augmented quality functions are described in FSAR Section 9.2.)

12. Breathing rate used to calculate the thyroid dose for the accident is 3.47×10^{-4} m^3/sec.

Prior to the steam generator tube rupture (SGTR) accident, it is assumed that the plant has been operating with simultaneous fuel defects and steam generator tube leakage for a period of time sufficient to establish equilibrium levels of radioactivity in primary and secondary coolant. The offsite and control room doses following a SGTR are analyzed considering both pre-accident and accident initiated iodine spikes. For the preaccident iodine spike, it is assumed that a reactor transient has occurred prior to SGTR and has raised the RCS iodine concentration to the allowed Technical Specification value of 50 μCi/g. For the accident-initiated iodine spike, 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 of 500 times greater than the normal equilibrium rate corresponding to the initial RCS iodine activity. For both of these iodine spike cases, the SGTR radiological analysis includes three primary sources of activity: (1) initial secondary side iodine activity, (2) RCS coolant activity released via primary to secondary steam generator tube leakage in the intact steam generator, and (3) RCS coolant activity carried over from the primary coolant via the ruptured steam generator tube.

The model for the activity available for release to the atmosphere from the ruptured and intact steam generators assumes that the release consists of the activity in the secondary coolant prior to the accident plus that activity leaking from the primary coolant through the SG tubes following the accident. The primary coolant activity after the accident is assumed to be composed of the pre-accident iodine spike activity or accident initiated iodine spike activity, plus the noble gases released due to 1% fuel defects. The leakage of primary coolant to the secondary side of the SG is assumed to continue at its initial rate of 0.35 gpm in the intact SG for the duration of the accident. A coincident loss of offsite power is assumed resulting in the loss of the condensers and the release of activity to the atmosphere through the main steam safety valves and the atmospheric steam relief valve from the intact steam generator. Eight hours after the accident, no further steam or activity is released to the environment.

A separate thermal hydraulic analysis was performed to determine the amount of reactor coolant transferred to the secondary side of the ruptured steam generator and the amount of steam released from the ruptured and intact steam generators to the atmosphere. This analysis was performed at an uprated power level of 1650 Mwt (Reference 5). Per this analysis the break flow through the ruptured steam generator will deliver 123,600 lbm of reactor coolant to the secondary side of the steam generator. None of the break flow is assumed to flash in the steam generator resulting in a direct release to the environment. The release to the environment is assumed to persist until 30 minutes after the initiation of the SGTR, at which time it is assumed that the operators have completed the actions necessary to terminate the break flow and the steam release from the ruptured steam generator. The amount of steam released from the ruptured steam generator during the 30 minute time period is calculated to be 74,000 lbm. A partition factor of 0.01, as defined in Standard Review Plan 15.6.3 (Reference 8), is applied to this steam release. No credit is taken for additional partitioning in the condenser prior to reactor trip. Both the break flow and steam releases are averaged over the 30 minute time interval.

The Westinghouse analysis for SGTR is comprised of 5 separate computer runs; a nominal RCS iodine activity case, a pre-accident iodine spike case, an accident initiated iodine spike case, an initial secondary coolant iodine case and a noble gas case. Each of the iodine cases model the releases to the environment from both the intact and ruptured steam generator using a partition factor of 0.01 on the steam releases. For each of these cases, except the initial secondary coolant iodine case, a transfer is modeled from the RCS to the steam generators based on primary to secondary leakage to the intact SG and the breakflow through the ruptured steam generator.

The activities in the steam generators and those released to the environment at the end of the 30 minute and 8 hour time intervals were evaluated. After 30 minutes, no further activity release from the ruptured steam generator was assumed. Primary to secondary leakage to the intact steam generator is terminated 8 hours after the accident.

The analysis specifies a total release time of 8 hours, for conservatism and to be more consistent with typical analysis values. The total steam releases from the intact and ruptured SGs are summarized below.

	Rate of Steam Release (g/min)	Mass of Steam Released (lbm)
Ruptured Steam Generator		
0 - 0.5 hr	1.12 E6	74,000
0.5 - 8 hr	0.0	0.0
Intact Steam Generator		
0 - 2 hr	6.28 E6	1.66 E6
2 - 8 hr	4.72 E5	3.74 E5

The noble gas case models a release directly from the RCS to the environment based on primary to secondary leakage to the intact SG and the breakflow through the ruptured SG assuming an RCS coolant activity corresponding to a fuel defect level of 1 percent.

Two release paths are assumed for the accident. For the source term in the RCS, releases are assumed to occur through the tubes in the steam generator with subsequent release out through the safety relief valves or the atmospheric steam dump valve for both the intact and ruptured steam generators. The release point for the safety relief valves or the atmospheric steam dump valve is assumed to be the vents associated with the valves. These valves exhaust to the environment through the top of the facade at an elevation of 170 feet. For both release points, atmospheric dispersion factors were calculated. The most limiting factor was then used for both release points to evaluate the dose consequences of the releases.

The thyroid dose conversion factors, breathing rates, and atmospheric dispersion factors used in the dose analysis are given in FSAR Table 14.1.8-3. The assumed core and coolant activities are given in FSAR Table 14.1.8-4.

Acceptance Criteria

The offsite dose limits for a SGTR accident with a pre-accident iodine spike are the guideline values of 10 CFR 100. These guideline values are 300 rem thyroid and 25 rem whole body. For a SGTR accident with an accident-initiated iodine spike, the acceptance criterion is a "small fraction" of the 10 CFR 100 guideline values; or 30 rem thyroid and 2.5 rem whole body (Reference 8). The criteria defined in the Standard Review Plan Section 6.4 are used for control room dose limits: 30 rem thyroid, 5 rem whole body, and 30 rem beta skin. (Reference 6)

Results/Conclusions

The thyroid and whole body doses at the site boundary and the low population zone are given in Table 14.2.4-1. The doses to the public as a result of a steam generator tube rupture are less than the permissible limits of 10 CFR Part 100. The control room whole body dose, thyroid dose, and beta skin dose are within the acceptance criteria. Independent assessments (Reference 7) of the onsite and offsite radiological doses have concluded that the loss of coolant accident (LOCA) is more limiting than the SGTR accident.

Multiple Tube Ruptures

A much larger dose, e.g., whole body dose of 25 rem at the exclusion radius, can only result from the rupture of sufficient steam generator tubes to cause fuel cladding failure.

Operating experience with steam generators of the type used in this plant has not shown significant numbers of single gross and immediate tube failures. Small leaks in a single tube which caused erosion type damage to adjacent tubes have been reported, but did not cause a rupture of the adjacent tubes. Thus, if a single tube failure were postulated, it is probable that adjacent tubes would not be damaged but any adjacent failure would be an erosion-caused leak rather than a sudden gross failure.

To perform a rigorous analysis of the flow dynamics of blowdown through multiple tube ruptures, one must understand and define mathematically the physical configuration of the ruptures. Because no reasonable mechanism exists for the multiple ruptures, it is instead just as meaningful to analyze the consequences of a pipe rupture, equivalent in terms of discharge rate to various multiples of the single tube discharge rate.

Such an analysis reveals that the core cooling system will prevent clad damage for break discharge rates equal to or smaller than that resulting from a broken pipe between 4 inches and 6 inches in diameter. The discharge rates which bracket the onset of cladding damage correspond to 18 and 40 times the discharge from a single severed steam generator tube. Actually, the ratio would be much larger owing to the fact that the discharge from a tube failure will be limited by the back pressure in the steam generator. Ultimately, the tube discharge would terminate when the reactor coolant system and the steam generator reached pressure equilibrium. The operator can initiate cooldown through the unaffected steam generator.

These conclusions are based on single-failure mode performances of the core cooling system. The core does not become uncovered by the calculated quiet level in those cases where cladding damage is found to be prevented.

The incredibility of multiple simultaneous tube failures is supported by the following reasoning:

1. At the maximum operating internal pressure the tube wall sees only about 1530 psi compared with a calculated bursting pressure in excess of 11,100 psi based on ultimate strength at design temperature.

2. The above margin applies to the longitudinal failure modes, induced by hoop stress. This failure mode is the least likely to cause propagation of failure tube-to-tube. An additional factor of two applies to ultimate pressure strength in the axial direction tending to resist double-ended failure (total factor of 14.6).

3. Failures induced by fretting, corrosion, erosion, or fatigue are of such a nature as to produce tell-tale leakage in substantial quantity while ample metal remains to prevent severance of the tube (a small fraction of the original tube wall section) as indicated by the margin derived in 2 above. Thus, any incipient failures that would develop to the point of severe leakage requiring a shutdown for plugging or repair, in accordance with Technical Specifications, would happen long before the large safety margin in pressure strength is lost.

References

1. L. C. Watson, A. R. Bancroft and C. W. Howlke, "Iodine Containment by Dousing in NPD-11" AECL-1130 Atomic Energy of Canada Limited, Chalk River, Ontario, October 27, 1960.

2. M. A. Styrikovich, O. I. Martynova, K. Ya. Katkovskaya, I. Ya. Dubrovskii, and I. N. Smirnova, "Transfer of Iodine from Aqueous Solutions to Saturated Vapor," Atomriaya Energiya, Vol. 17, No. 1, pp. 45-49, July 1964.

3. O. J. Mendler, "The Effect of Steam Generator Tube Uncovery on Radioiodine Release," WCAP-13132, January 1992.

4. R. C. Jones, U.S. NRC, Letter to L. A. Walsh, WOG, "Westinghouse Owners Group - Steam Generator Tube Uncovery Issue," March 10, 1993, Attachment toWOG-93-066 dated March 31, 1993.

5. WE letter to NRC, NPL 97-0144, "Supplement to Technical Specifications Change Requests 188 and 189", dated April 2, 1997.

6. WE letter to NRC, VPNPD-97-009, "Supplement to Technical Specifications Change Requests 188 and 189", dated January 16, 1997.

7. NRC Safety Evaluation Report (SER) dated July 1, 1997, "Issuance of Amendments for Technical Specification Change Requests 188 and 189".

8. NUREG-0800, Standard Review Plan, Section 15.6.3, "Steam Generator Tube Rupture."

9. NPL 94-0113, "Evaluation of Main Steam Safety Valves Blowdown on SGTR Analysis, Steam Generator Replacement, Point Beach Nuclear Plant", March 18, 1994.

10. PBNP 50.59 Evaluation EVAL 2001-004, "Mixed Bed Allowable Flow Increase and SGTR/MSLB Dose Consequence Revision," November 26, 2001.

11. NRC Safety Evaluation Report (SER) dated July 9, 1997, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 174 and 178 to Facility Operating License Nos. DPR-24 and DPR-27."

12. Amendment Nos. 223 and 229 to Renewed Facility Operating Licenses DPR-24 and. DPR-27 respectively, for the Point Beach Nuclear Plant, Units 1 and 2, dated August 22, 2006.

TABLE 14.2.4-1
THYROID DOSES AND WHOLE BODY DOSES
STEAM GENERATOR TUBE RUPTURE ACCIDENT

A. With Pre-Accident Iodine Spike

0 - 2 hr Dose At Site Boundary (Rem)		0 - 8 hr Dose At LPZ (Rem)	
Thyroid	Whole Body	Thyroid	Whole Body
3.5	0.1	0.2	6×10^{-3}

B. With Accident-Initiated Iodine Spike

0 - 2 hr Dose At Site Boundary (Rem)		0 - 8 hr Dose At LPZ (Rem)	
Thyroid	Whole Body	Thyroid	Whole Body
2.3	0.1	0.2	6×10^{-3}

FIGURE 14.2.4-1
EQUILIBRIUM IODINE CONCENTRATION IN STEAM GENERATOR WATER
vs.
PRIMARY TO SECONDARY LEAK RATE

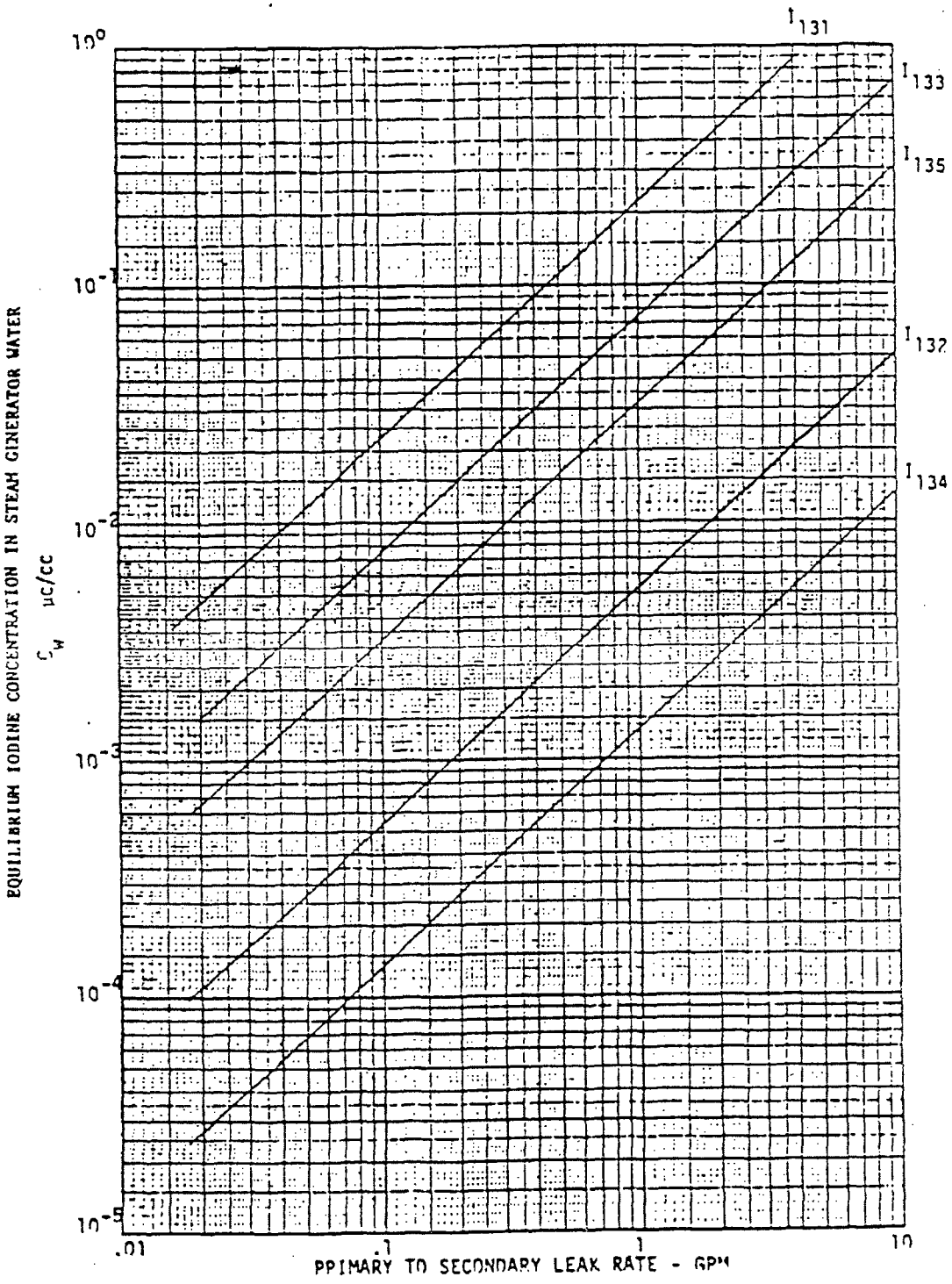


FIGURE 14.2.4-1

FIGURE 14.2.4-2
EQUILIBRIUM IODINE CONCENTRATION IN BLOWDOWN TANK WATER
vs.
PRIMARY TO SECONDARY LEAK RATE

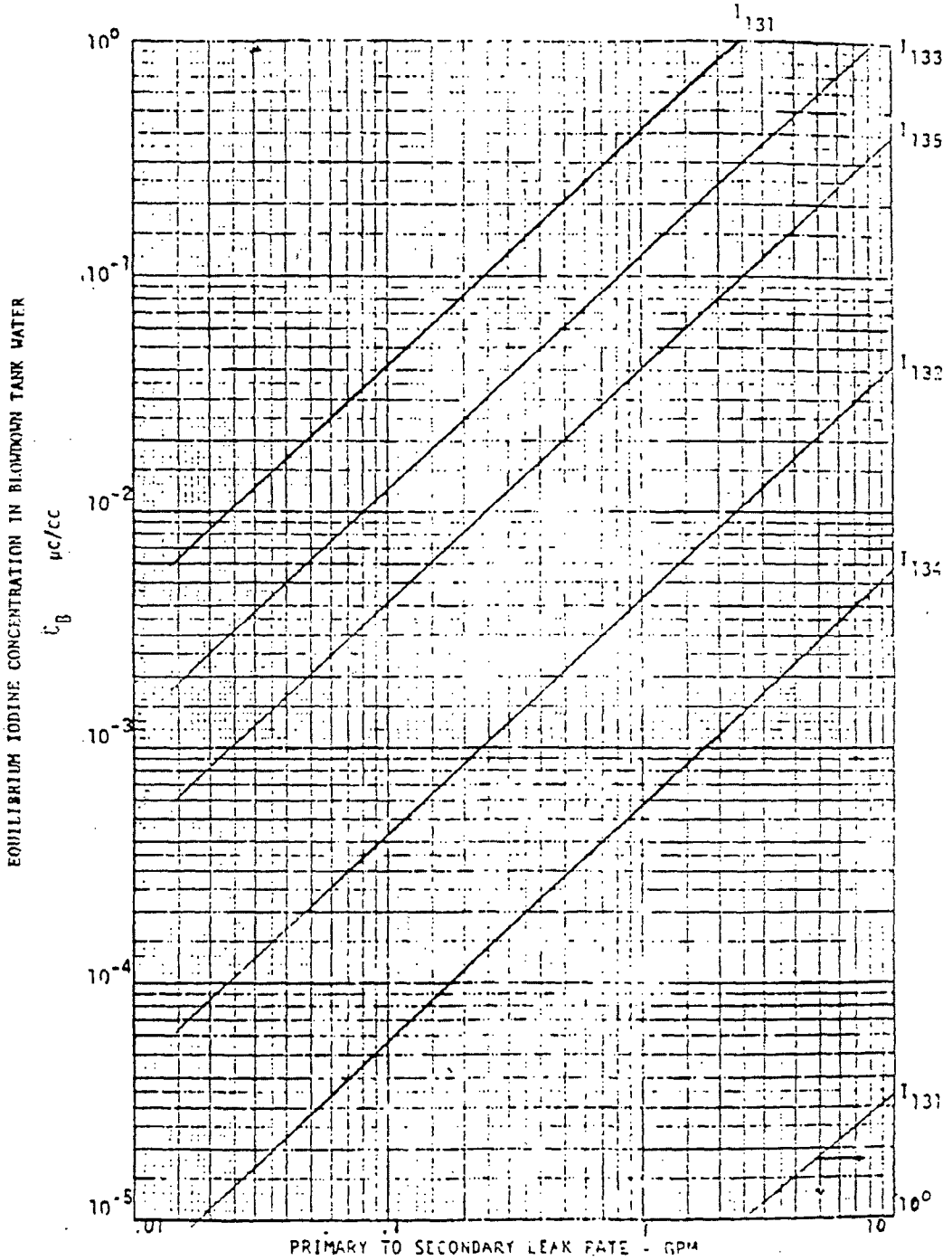


FIGURE 14.2.4-2

FIGURE 14.2.4-3
EQUILIBRIUM IODINE CONCENTRATION IN CONDENSER AIR-STEAM MIXTURE
vs.
PRIMARY TO SECONDARY LEAK RATE

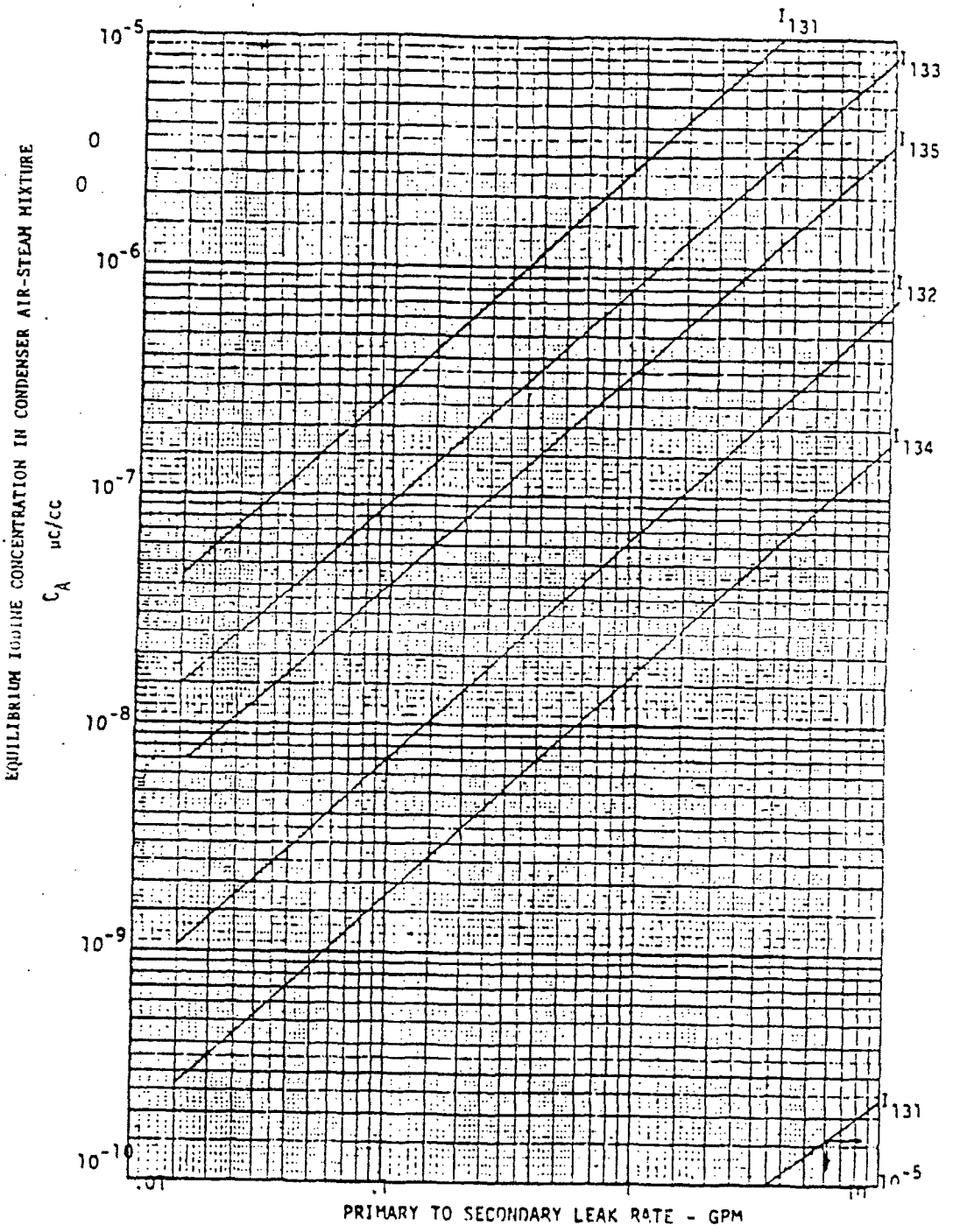


FIGURE 14.2.4-3

FIGURE 14.2.4-4
EQUILIBRIUM MOBILE GAS CONCENTRATION IN CONDENSER AIR-STEAM MIXTURE
vs.
PRIMARY TO SECONDARY LEAK RATE

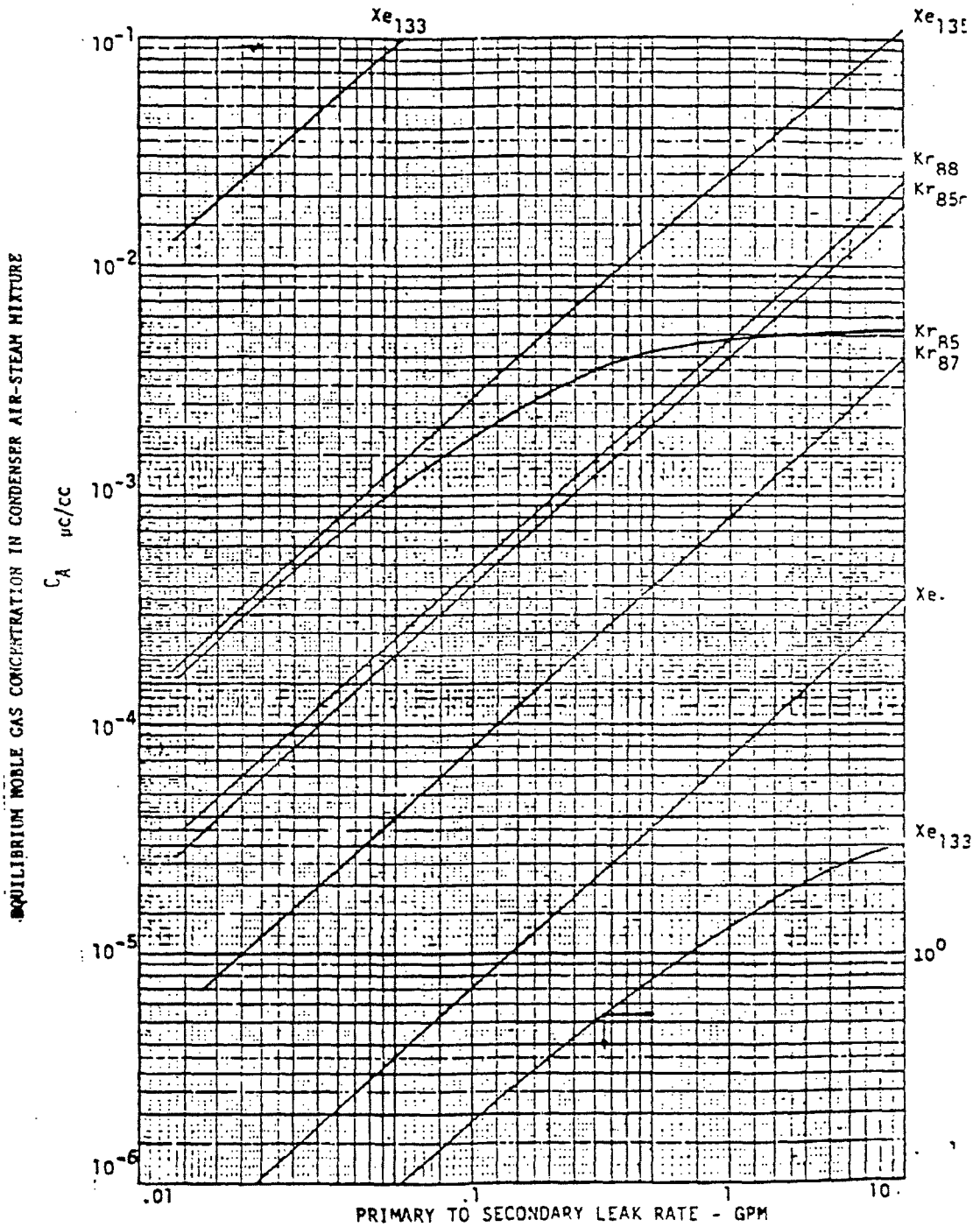


FIGURE 14.2.4-4

14.2.5 Rupture of a Steam Pipe

A. Core Power and Reactor Coolant System Transient

A rupture of a steam pipe is assumed to include any accident which results in an uncontrolled steam release from a steam generator. The release can occur due to a break in a pipe line or due to a valve malfunction. The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant temperature and pressure. With a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive control rod is assumed stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power even with the remaining control rods inserted. A return to power following a steam pipe rupture is a potential problem only because of the high hot channel factors which may exist when the most reactive rod is assumed stuck in its fully withdrawn position. Assuming the most pessimistic combination of circumstances which could lead to power generation following a steam line break, the core is ultimately shut down by the boric acid in the Safety Injection System.

The analysis of a steam pipe rupture is performed to demonstrate that:

1. With a stuck rod and minimum engineered safety features, the core remains in place and essentially intact so as not to impair effective cooling of the core.

2. With no stuck rod and all equipment operating at design capacity, insignificant cladding rupture occurs.

Although DNB and possible cladding perforation (no cladding melting or zirconium-water reaction) following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive rod stuck in its fully withdrawn position. The following systems provide the necessary protection against a steam pipe rupture:

1. Safety Injection System actuation on:

    a. Two out of three pressurizer low pressure signals.

    b. Two out of three low pressure signals in any steam line.

    c. Two out of three high containment pressure signals.

2. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring upon actuation of the Safety Injection System.

3. Redundant isolation of the main feedwater lines. Sustained high feedwater flow would cause additional cooldown, thus, in addition to the normal control action which will close the main feedwater valves, any safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater pump discharge valves. Additional isolation is provided by tripping the condensate and heater drain tank pumps on a high containment pressure safety injection signal to help prevent over-pressurization of the containment for ruptures inside containment.

4. Trip of the fast acting steam line isolation valves (designed to close in less than 5 seconds with low flow, upon receipt of a CLOSE signal) on:

   a. One out of the two high steam flow signals in that steam line in coincidence with any safety injection signal. (Dual set points are provided, with the lower set point used in coincidence with two out of four indications of low reactor coolant average temperature.)

   b. Two out of three high - high containment pressure signals.

Each steam line has a fast closing isolation valve and a check valve. These four valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For example, for a break upstream of the isolation valve in one line, closure of either the check valve in that line or the isolation valve in the other line will prevent blowdown of the other steam generator.

Steam flow is measured by monitoring dynamic head in nozzles inside the steam pipes. The nozzles (16 in. I.D. vs. a pipe diameter of 28 in. I.D.) are located inside the containment near the steam generator. The Unit 1 and Unit 2 steam generators contain a steam nozzle flow limiting device which is designed to limit the steam generator depressurization rate by restricting the steam flow during any postulated steam line break accident.

Method of Analysis

The analysis of the steam pipe rupture has been performed to determine:

1. The core heat flux and reactor coolant system temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN code has been used.

2. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital computer code, THINC, has been used to determine if DNB occurs for the core conditions computed in (1) above.

3. The offsite consequences of the steam line break accident which include consideration of the additional secondary loop activity resulting from a steam generator tube leak prior to the accident.

4. The onsite consequences (e.g., control room habitability). These analyses are described in general terms in this section.

The following assumptions are made:

1. A 2.77% shutdown reactivity from the rods at no load conditions. This is the end of life design value including design margins with the most reactive rod stuck in its fully withdrawn position. Operation of the RCCA banks is restricted in accordance with the Technical Specifications such that the main steam line break analysis remains bounding.

2. The negative moderator temperature coefficient corresponding to the end of life core with all but the most reactive rod inserted. The variation of the coefficient with temperature and pressure has been included. In computing the power generation following a steam line break, the local reactivity feedback from the high neutron flux in the region of the core near the stuck control rod has been included in the overall reactivity balance. The local reactivity feedback is composed of Doppler reactivity from the high fuel temperatures near the stuck control rod and moderator feedback from the high water enthalpy near the stuck rod. For the cases analyzed where steam generation occurs in the high flux regions of the core the effect of void formation on the reactivity has been included. The effect of power generation in the core on overall reactivity is a function of the core temperature, pressure, and flow and thus is different for each case studied. The curves assume end of life core conditions with all rods in except the most reactive rod which is assumed stuck in its fully withdrawn position.

3. Minimum capability for injection of 2,000 ppm boric acid solution corresponding to the most restrictive single failure in the safety injection system. The emergency core cooling system consists of three systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, and 3) the high head safety injection system. Both the accumulators and the high head safety injection are modeled for the steamline break accident analysis. The boric acid solution of the high head safety injection and the accumulators is 2000 ppm.

   The modeling of the safety injection system in LOFTRAN is described in Reference 1. The flow corresponds to that delivered by one safety injection pump delivering its full flow to both RCS cold legs. The accumulators are modeled to begin injection when the cold leg pressure drops to 700 psia.

   For cases where offsite power is available, the sequence of events in the safety injection system is the following: After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high head safety injection pump starts. Ten seconds later, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing unborated water is swept into the core before the 2,000 ppm borated water reaches the core. This delay, described above, is included in the modeling.

In cases where offsite power is not available, maximum delay times are considered to account for SI signal processing (2 seconds), diesel generator start to full speed (15 seconds), and SI pump start to full speed (10 seconds), for a total delay of 27 seconds, is assumed in the analysis. This delay time is also applied to the cases where offsite power is available for conservatism.

4. Design value of the steam generator heat transfer coefficient including allowance for fouling factor.

5. Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck RCCA. The power peaking factors account for the effect of the local void in the region of the stuck RCCA during return to power phase following the steamline break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck RCCA. The power peaking factors depend upon the core power, temperature, pressure, and flow, and thus are different for each case studied.

6. Six initial plant conditions have been considered in determining the core power and reactor coolant system transient.

    a. Complete severance of a pipe outside the containment, for Unit 1, at initial no load conditions, with offsite power available.

    b. Complete severance of a pipe outside the containment for Unit 2, at initial no load conditions, with offsite power available.

    c. Case (a) above with only one loop in service.

    d. Case (b) above with only one loop in service.

    e. Case (a) above with loss of offsite power simultaneous with break.

    f. Case (b) above with loss of offsite power simultaneous with break.

The cases above assume initial hot shutdown conditions with the rods inserted (except for one stuck rod) at time zero. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when the power level reaches a trip point.

Following a trip at power, the reactor coolant system contains more stored energy than at no load, the average coolant temperature is higher than at no load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no load conditions of reactor coolant system temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analyses which assume no load conditions at time zero.

Results

The results presented are a conservative indication of the events which would occur assuming a steam line rupture. The worst case assumes that all of the following occur simultaneously.

1. Minimum shutdown reactivity margin equal to 2.77%.

2. The most negative moderator temperature coefficient for the rodded core at end of life.

3. The rod having the most reactivity stuck in its fully withdrawn position.

4. One safety injection pump fails to function as designed.

Figures 14.2.5-1through 14.2.5-6 show the reactor coolant system transient and core heat flux following a steam pipe rupture for each of the cases considered. A maximum break area of 1.4 ft^2 is assumed which is the effective cross-sectional area of the steam nozzle flow limiting devices on the Unit 1 and Unit 2 steam generators. For the offsite power available cases, boron injection from both the passive accumulators and one high-head safety injection pump prevents the core from returning to power above an initial core heat flux peak of 3.1% (two-loop) and 2.4% (one-loop). For the loss of offsite power cases, boron injection is provided only by one high-head safety injection pump since RCS pressure never reached the accumulator setpoint (700 psia). For this case, peak core heat flux (3.1%) occurs after criticality is reached. The transient shown assumes the rods inserted at time 0 (with one rod stuck in its fully withdrawn position) and steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs, the initiation of safety injection by low steam line pressure will trip the reactor. Steam release from at least one steam generator will be prevented by either the check valve or by automatic trip of the fast acting isolation valve in the steam line by the high steam flow signal in coincidence with the safety injection signal. Even with the failure of one valve, steam release is limited to no more than five seconds for one steam generator while the second generator blows down. (The steam line isolation valves are designed to be fully closed in less than five seconds following receipt of closure signal under low flow conditions. With the high flow existing during a steam line rupture, the valves will close considerably faster.)

A sequence of events for each steamline break case considered is provided in Table 14.2.5-2. The value of peak heat flux for each case and the time of occurrence are also provided in Table 14.2.5-2 (Reference 16).

As the cooldown effects resulting from a credible steamline break (e.g., a malfunction of an Atmospheric Dump Valve) are less severe than the cooldown effects that would result from a hypothetical steamline break (i.e., the double-ended rupture), the results of the credible steamline break are bounded by the analysis results of the hypothetical steamline break. In 1990, as described in WCAP-12602 (Ref. 11), the credible and hypothetical steamline break scenarios were analyzed in support of the SI system boron concentration reduction. For the reanalysis of the credible steamline break scenario, the conservative acceptance criterion of "no return to criticality" applied by Westinghouse was dropped and replaced by the same criterion applied in the hypothetical steamline break analysis, i.e., the analysis results were shown to meet the radiation release limits set forth in 10CFR Part 20 by demonstrating that the DNB design basis was met. Additionally, safety injection was conservatively assumed to be initiated by a low pressurizer signal in the credible steamline break analysis, and a low steamline pressure signal in the hypothetical steamline break analysis. A comparison between the results of the credible and hypothetical scenarios clearly showed that the hypothetical steamline break is limiting with respect to minimum DNBR. Based on this, it was no longer necessary to analyze the credible steamline break for Point Beach.

B. Radiological Consequences

The complete severance of a main steamline outside containment is assumed to occur. The affected SG will rapidly depressurize and release to the outside atmosphere the radioiodines initially contained in the secondary coolant and the radioiodines which are transferred from the primary coolant through SG tube leakage. 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 or the main steam safety valves. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite doses resulting from the release. (Reference 9)

The onsite release and radiological analyses were conducted in support of Unit 2 Steam Generator replacement. The offsite consequences were generally accepted by the NRC (Reference 8) and further modified by Reference 12 to correct non-conservative input parameters used to calculate the accident-initiated iodine spike. The analyses of the control room habitability submitted in support of the Unit 2 Steam Generator replacement were subsequently identified for further review. Therefore a quantifiable description of the control room habitability assessments are not provided herein. The general results are provided in that independent assessments (Reference 8) of the onsite and offsite radiological doses have concluded that the loss of coolant accident (LOCA) is more limiting than the MSLB accident.

The analysis of the Main Steam Line Break (MSLB) offsite radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan (Reference 10). For the pre-accident iodine spike, it is assumed that a reactor transient has occurred prior to the MSLB and has raised the RCS iodine concentration to the allowed Technical Specification value of 50 μCi/gm of dose equivalent (DE) I-131. For the accident-initiated iodine spike, the reactor trip associated with the MSLB creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value of 500 times greater than the release rate corresponding to the maximum proposed equilibrium RCS Technical Specification concentration of 0.8 μCi/gm of DE I-131. The duration of the accident-initiated iodine spike is assumed to be 1.6 hours. (Reference 9)

The noble gas activity concentration in the RCS at the time of the accident 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 of the MSLB is assumed to be equivalent to the Technical Specification limit of 1.0 μCi/gm of DE I-131. The amount of primary-to-secondary SG tube leakage in each of the two SGs is assumed to be equal to 500 gallons/day (0.35gpm) or 1000 gallons/day (0.70 gpm) total. Technical Specifications provide a basis for this assumption by the establishment of: 1) a primary to secondary operational leakage limit of 150 gallons/day per Steam Generator, and 2) a Steam Generator Program which includes structural integrity and accident induced leakage performance criteria. (Reference 15) 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 ruptured main steamline is assumed to boil dry within the initial half hour following the MSLB. 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 SG tube leaks is assumed to be released directly to the environment with no credit taken for iodine retention in the SG.

The following assumptions and parameters are also used in the analyses.

1. The activities in the primary and secondary systems are at equilibrium prior to the accidents.

2. The iodine partition factor (Amount iodine/unit vol. gas)/ (Amount iodine/unit vol. liquid) is assumed to be 0.01 in the intact steam generator (Reference 9).

3. Auxiliary feedwater is available during the accidents.

4. The atmospheric dispersion factors (X/Q) at site boundary and at the boundary of low population zone for 0 - 2 hour and 2 - 8 hour periods following the accidents are given in Table 14.1.8-3.

5. Breathing rate used to calculate the thyroid dose for the accidents is 3.47×10^{-4} m^3/sec.

6. The postulated rupture of the main steam line occurs outside of the containment.

7. Following the accident the primary coolant pressure is at 2235 psig for the first two hours. And from 2 - 8 hr the primary coolant pressure is reduced from 2235 psig to 335 psig.

8. Steam release to atmosphere is used to cool the primary coolant prior to switching to the residual heat removal system.

9. Eight hours after the accident the residual heat removal system is assumed to be placed in service and there are no further steam releases to the atmosphere from the secondary system. (Note that this is an analysis assumption in the approved radiological analysis that terminates steam releases. The RHR system safety-related and augmented quality functions are described in FSAR Section 9.2.)

10. The core and coolant activities used in the radiological analyses are given in Table 14.1.8-4.

Acceptance Criteria

The offsite dose limits for a MSLB accident with a pre-accident iodine spike are the guideline values of 10 CFR 100. These guideline values are 300 rem thyroid and 25 rem whole body. For a MSLB accident with an accident-initiated iodine spike, the acceptance criterion is a "small fraction" of the 10 CFR 100 guideline values; or 30 rem thyroid and 2.5 rem whole body. The criteria defined in the Standard Review Plan Section 6.4 are used for control room dose limits: 30 rem thyroid, 5 rem whole body, and 30 rem beta skin.

C. Containment Response Analysis

An analysis is performed to predict the pressure and temperature response of the containment atmosphere to a main steamline break inside of containment. The steamline break is postulated as a full double-ended rupture (DER) of the steamline immediately downstream of the steam generator integral flow restrictor. The blowdown from the faulted steam generator is limited by the 1.4 ft^2 integral flow restrictor. The steamline non-return valve limits the reverse break flow to the steam in the steamline between the break and the non-return valve.

The analysis includes a single failure of the feedwater-regulating valve (FRV) on the faulted loop. Although the FRV is an air-operated valve that would typically fail in a closed position, it is assumed to fail in the fully open position. The open FRV allows additional main feedwater to be pumped into the faulted steam generator until the main feedwater pumps coast down after they are tripped on an SI signal. Furthermore, there is a large unisolable feedline volume between the faulted steam generator and the main feedwater pump discharge valves. The hot water in the feedline will flash and enter the faulted steam generator when the feedwater becomes saturated due to the depressurization of the system.

The only case considered in this analysis is the full DER from 102% power with a single failure of the FRV. This was determined to be the case resulting in the highest containment pressure based on sensitivity analyses performed for a future power uprate (Reference 5). This case is limiting because of the relatively high energy transfer to the faulted steam generator and the early flashing of the large quantity of feedwater in the unisolable feedline.

Method of Analysis

The analysis consists of the calculation of the mass and energy releases from the steamline break and the calculation of the containment pressure and temperature response. The methods and assumptions of these calculations are summarized below.

Mass and Energy Release Calculation

WCAP-8822, "Mass and Energy Releases Following a Steam Line Rupture" (Reference 2) forms the basis for the assumptions and models used in the calculation of the mass and energy releases resulting from a steamline rupture. The steamline break mass and energy releases are generated using the NRC-approved LOFTRAN code (Reference 1). The Westinghouse steamline break mass and energy release methodology using LOFTRAN was approved by the NRC and is documented in Supplement 2 to WCAP-8822 (Reference 3).

The major inputs and assumptions affecting the mass and energy releases to containment are summarized below (Reference 4).

- The initial NSSS power level is 102% of 1524.5 MWt.
- The initial RCS average temperature is 575.6°F, which includes a +5.6°F uncertainty.
- The core nuclear power transient due to the cooldown following the steamline rupture is based on end-of-core life conditions with the most reactive control rod stuck out of the core. The credited shutdown margin is 3.1%Δk.
- Two sources of latent energy to the reactor coolant system are modeled: the reactor vessel and primary system thick metal, and the fluid inventory in the intact steam generator.
- Offsite power is assumed to remain available. The largest effect of this assumption is the continued operation of the reactor coolant pumps, which maintains a high heat transfer rate to the steam generators.
- Minimum flowrates are modeled from ECCS injection, to conservatively minimize the amount of boron that provides negative reactivity feedback. The flowrates correspond to a single train of ECCS with 10% pump head curve degradation.
- A high initial steam generator mass is assumed. The initial level corresponds to 64% NRS + 4% uncertainty.
- The calculation of secondary side break flow is based on the Moody critical flow correlation with fL/D=0.

- The main feedwater modeling accounts for an increase from the initial flowrate due to the depressurization of the faulted steam generator and the opening of the FRV in response to the increased steam flow. Main feedwater pumped flow is terminated by the trip of the main feedwater pumps and corresponding pump coastdown.

- Feedline flashing occurs when saturated conditions are reached in the 1198 ft^3 unisolable volume between the faulted steam generator and the main feedwater pump discharge valves. The homogenous flashing model in LOFTRAN was used, but with two separate volumes to account for the water that was heated to 430°F by the feedwater heaters, and the water upstream of the heaters at a temperature of 350°F.

- Maximum flowrates of auxiliary feedwater (AFW) were assumed, with the AFW start conservatively modeled at the time of the SI signal, with no delay. AFW is assumed to be manually re-aligned at 600 seconds to prevent further water addition to the faulted steam generator (See Results Section for additional discussion).

- The steam in the unisolable volume of 1650 ft^3 between the faulted steam generator and the steamline non-return check valve comprises the reverse flow from the break.

- The break effluent is assumed to be dry, saturated steam throughout most of the transient. However, when a large double-ended break first occurs, it is expected that there will be a significant quantity of liquid in the break effluent. A conservative amount of liquid entrainment is assumed to occur in the beginning of the steam generator blowdown phase of the accident. The break effluent is assumed to return to all vapor within the first 25 seconds.

- The containment backpressure is modeled within LOFTRAN as a function of time to calculate break flow. Modeling the backpressure causes the transition from critical to non-critical flow to occur earlier in the transient, reducing the break flow rate. Modeling the containment backpressure makes the interface between the mass & energy release and the containment response analysis consistent and more physically realistic.

- The time to tube uncovery was modeled in the same manner as was used in Reference 3 for "predicted tube uncovery" cases. This affects the total amount of heat transfer to the secondary side, and the possible generation of superheated steam.

Containment Response Calculation

The COCO computer code (Reference 6) is used to calculate the containment pressure and temperature transient response following the postulated steamline break accident inside containment.

The initial conditions (Table 14.2.5-3) are selected to maximize the containment pressure response. The initial pressure has a direct relationship on the peak containment pressure, and thus is maximized. The initial temperature is maximized because the steady-state temperature of the containment heat sinks are assumed to be the same as the containment air temperature. The higher initial heat sink temperature causes them to be less effective in removing heat. The initial humidity is conservative when it is assumed to be low, since this maximizes the amount of air initially in the containment.

Two trains of containment fan coolers (four coolers) and two trains of containment spray are credited. This is because a limiting single failure of the FRV has already been modeled in the mass and energy release calculation. The containment fan cooler heat removal performance was reduced 25% from the nominal value. A conservatively high temperature has been assumed as the temperature of the spray water. The containment spray pump flow performance includes a conservative pump head reduction of 10%.

Finally, the heat transfer through, and heat storage in, interior and exterior walls of the containment structure are considered. Structural heat sinks, consisting of steel and concrete, are modeled as slabs having specific areas and layers of varying thickness. The initial temperature of the structural heat sinks is assumed to be the initial containment air temperature of 120°F.

Results

The containment pressure and containment temperature transients are shown in Figures 14.2.5-7 and 14.2.5-8. The peak containment pressure of 59.8 psig is reached at 276 seconds, which is below the 60 psig containment design pressure. The peak containment temperature of 285°F is also reached at 276 seconds, which is below the containment design temperature of 286°F (see FSAR Section 5.1).

Both containment pressure and temperature trend consistently downward after peaking. This is due to heat removal by both active systems and passive heat sinks exceeding heat introduction from the break. At 600 seconds, the rate of pressure and temperature drop increases when AFW flow is isolated to the faulted stream generator. It is apparent from Figures 14.2.5-7 and 14.2.5-8 however, that the isolation of AFW at 600 seconds does not affect the peak containment pressure and temperature experienced earlier in the transient because both parameters are already decreasing before AFW isolation occurs. Therefore, while the manual isolation of AFW is an input assumed by the analysis, the results of the analysis show that the manual action is not necessary to ensure that containment integrity is maintained.

Note that the MSLB containment response analysis has been reviewed and approved by the NRC (Reference 7).

Conclusions

A DNB analysis has been performed. It was found that all cases have a minimum DNBR greater than the limit value.

The analysis has shown that the criteria stated in Section 14.2.5 are satisfied. Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis, in fact, shows that the DNB design basis is met as stated in Section 3.2.

No significant exposure to the public would result from a rupture of a steam pipe.

The containment pressure and temperature responses to a MSLB inside of containment remain below the containment design pressure and temperature.

References

1. Burnett, T.W.T., et al. "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.

2. Land, R.E., "Mass and Energy Releases Following a Steam Line Rupture," WCAP-8822 (Proprietary), WCAP-8860 (Non-Proprietary), September 1976.

3. Butler, J.C., "Mass and Energy Releases Following a Steam Line Rupture, Supplement 2 - Impact of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture for Dry and Subatmospheric Containment Designs," WCAP-8822-S2-P-A (Proprietary), WCAP-8860-S2-A (Non-Proprietary), September 1986.

4. WEP-01-060, "Containment Response to Steamline Break at 1524.5 MWt NSSS Power-Final Report," October 29, 2001.

5. Ohkawa, D.K., "Wisconsin Electric Power Company Point Beach Nuclear Plant, Units 1 and 2 Steamline Break and Containment Integrity Analysis," WCAP-15153 (Proprietary), December 1998.

6. "Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary), WCAP-8326 (Non-Proprietary), July 1974.

7. NRC Safety Evaluation Report (SER) dated November 26, 2002, "Issuance of Amendments RE: Change of Containment Maximum Pressure Technical Specification Limit," Amendment Nos. 206/211.

8. NRC Safety Evaluation Report (SER) dated July 1, 1997, "Issuance of Amendments for Technical Specification Change Requests 188 and 189."

9. WE letter to NRC, VPNPD-97-009, "Supplement to Technical Specifications Change Requests 188 and 189," dated January 16, 1997.

10. NUREG-0800, Standard Review Plan 15.1.5, Appendix A, "Radiological Consequences of Main Steam Line Failures Outside of a PWR," Rev. 2, July 1981.

11. WCAP-12602, "Report for the Reduction of SI System Boron Concentration," September 1990.

12. PBNP 50.59 Evaluation EVAL 2001-004, "Mixed Bed Allowable Flow Increase and SGTR/MSLB Dose Consequence Revision," November 26, 2001.

13. Westinghouse Letter WEP-05-209, "High Head Safety Injection Pump Spin-Up Time Accident Analysis Evaluation," July 16, 2005.

14. Westinghouse Letter WEP-05-317," AFW Operator Action Time for Point Beach Steamline Break," November 15, 2005

15. Amendment Nos. 223 and 229 to Renewed Facility Operating License Nos. DPR-24 and DPR-27, respectively for the Point Beach Nuclear Plant, Units 1 and 2, dated August 22, 2006.

16. Westinghouse Calculation Note CN-TA-96-075, Revision 1, "Hot Zero Power Steamline Break Assessment for a Reduced Zero Power Main Feedwater Temperature."

TABLE 14.2.5-1
THYROID DOSES AND WHOLE BODY DOSES
MAIN STEAMLINE BREAK ACCIDENT

Site Boundary (0 - 2 hr)	Dose (Rem)
Thyroid Dose (Accident-Induced Spike)	9.7
Thyroid Dose (Pre-Accident Spike)	8.3
Whole Body Dose	0.04

Low Population Zone (0 - 8 hr)	Dose (Rem)
Thyroid Dose (Accident-Induced Spike)	1.32
Thyroid Dose (Pre-Accident Spike)	0.7
Whole Body Dose	0.005

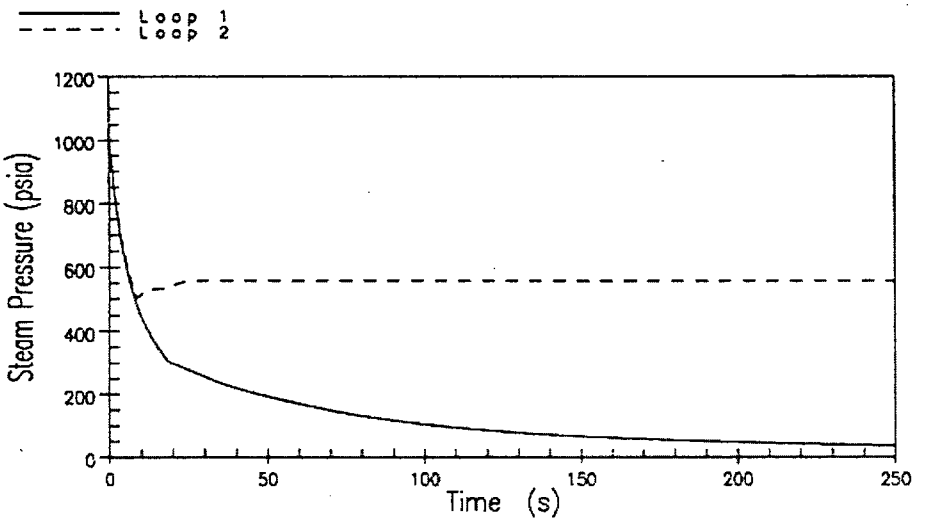
TABLE 14.2.5-2
RUPTURE OF A STEAM PIPE
ANALYSIS ASSUMPTIONS AND SEQUENCE OF EVENTS

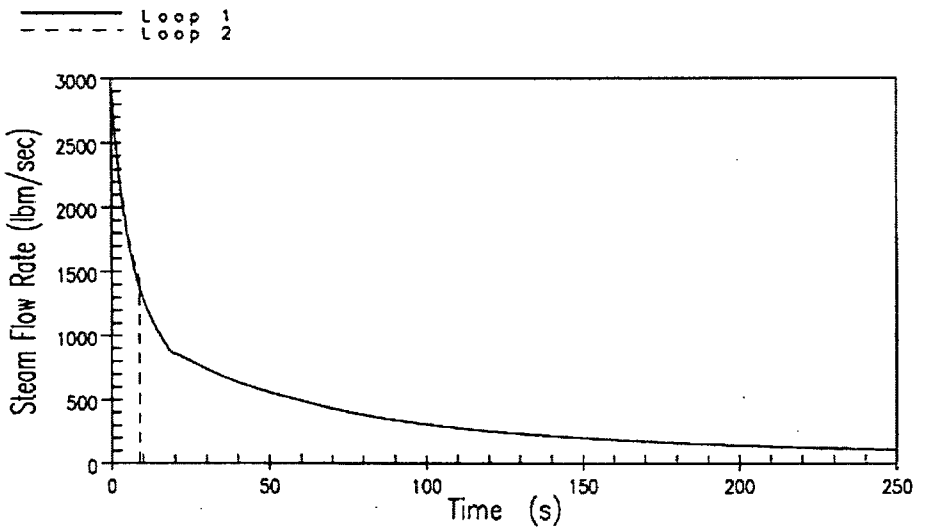
PBNP Unit Affected	Unit1	Unit 1	Unit 2	Unit 2	Unit 2	Unit 1
Steam Generator Model	44F	44F	Delta-47	Delta-47	Delta-47	44F
Number of Loops in Service	2	1	2	1	2	2
Initial shutdown margin, %Δk	2.77	2.77	2.77	2.77	2.77	2.77
Offsite Power Available	Yes	Yes	Yes	Yes	No	No
Rupture occurs in main steamline, sec	0.0	0.0	0.0	0.0	0.0	0.0
High-High steam flow setpoint reached, sec	0.1	0.1	0.1	0.1	0.1	0.1
Low steam pressure SI setpoint reached, sec	1.4	NA	1.4	NA	1.4	1.4
Steamline isolation occurs, sec	8.4	NA	8.4	NA	8.4	8.4
Low pressurizer pressure SI setpoint reached, sec	13.3	20.3	13.1	20.0	14.5	14.2
Feedwater isolation occurs, sec	18.4	37.3	18.4	37.0	18.4	18.4
Safety injection pump at full speed, sec	28.4	47.3	28.4	47.0	28.4	28.4
Core returns to criticality, sec	~92	~94	~116	NA	~84	~86
Boron reaches core, sec	~42	~62	~42	~62	~46	~46
Time of maximum core heat flux, sec	~10	~14	~10	~14	~232	~242
Maximum core heat flux, fraction of nominal	0.031	0.024	0.031	0.024	0.030	0.031

TABLE 14.2.5-3
COCO Model Inputs
MSLB Containment Response Analysis

Input	Value
RWST water temperature for containment sprays (°F)	100
Initial containment temperature (°F)	120
Initial containment pressure (psia)	16.7
Initial relative humidity (%)	20
Net free volume (ft^3)	1.0×10^6

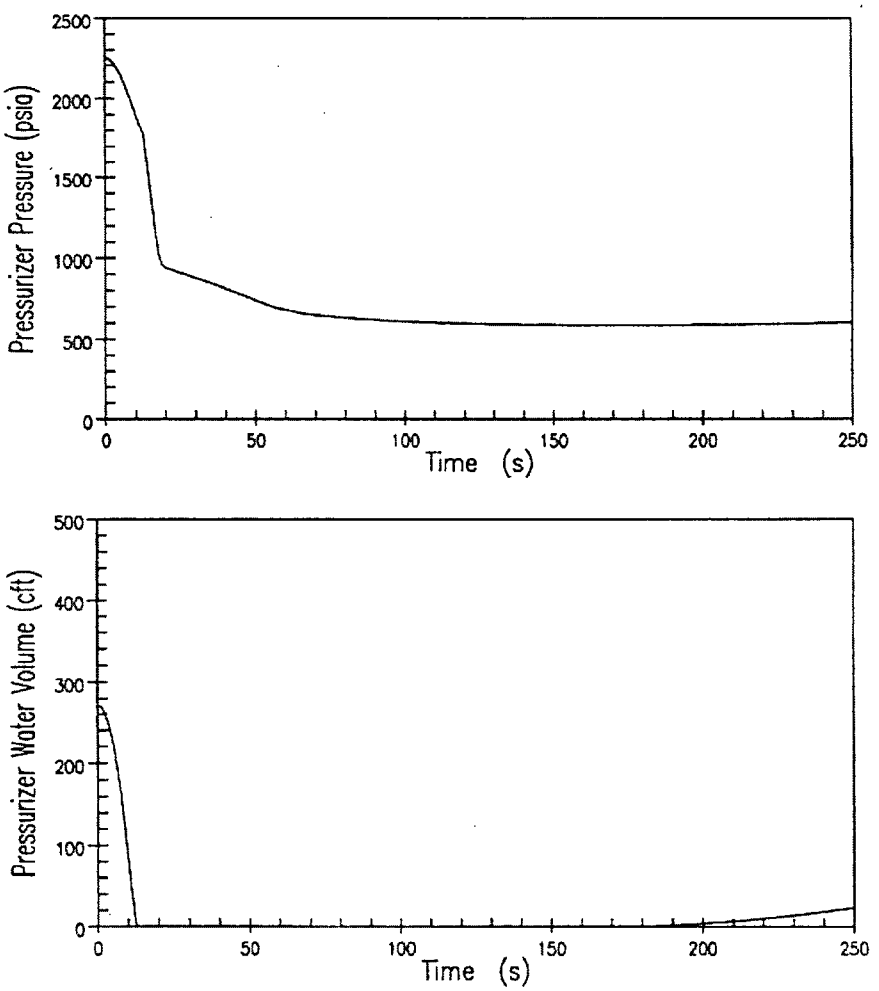
FIGURE 14.2.5-1
Sheet 1 of 4
RUPTURE OF A STEAM PIPE
UNIT 1 (two loops in service) WITH OFFSITE POWER





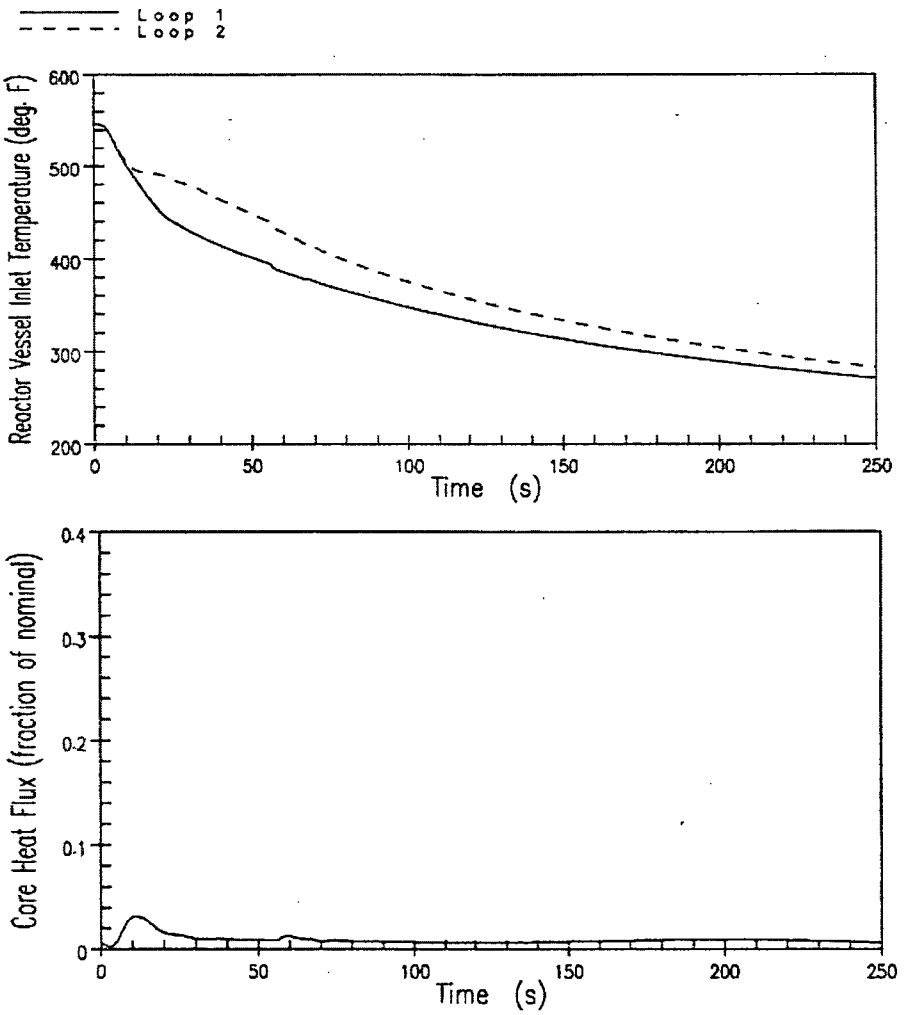
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 1 (two loops in service) With offsite power Figure 14.2.5-1 (sheet 1 of 4)

FIGURE 14.2.5-1
Sheet 2 of 4
RUPTURE OF A STEAM PIPE
UNIT 1 (two loops in service) WITH OFFSITE POWER



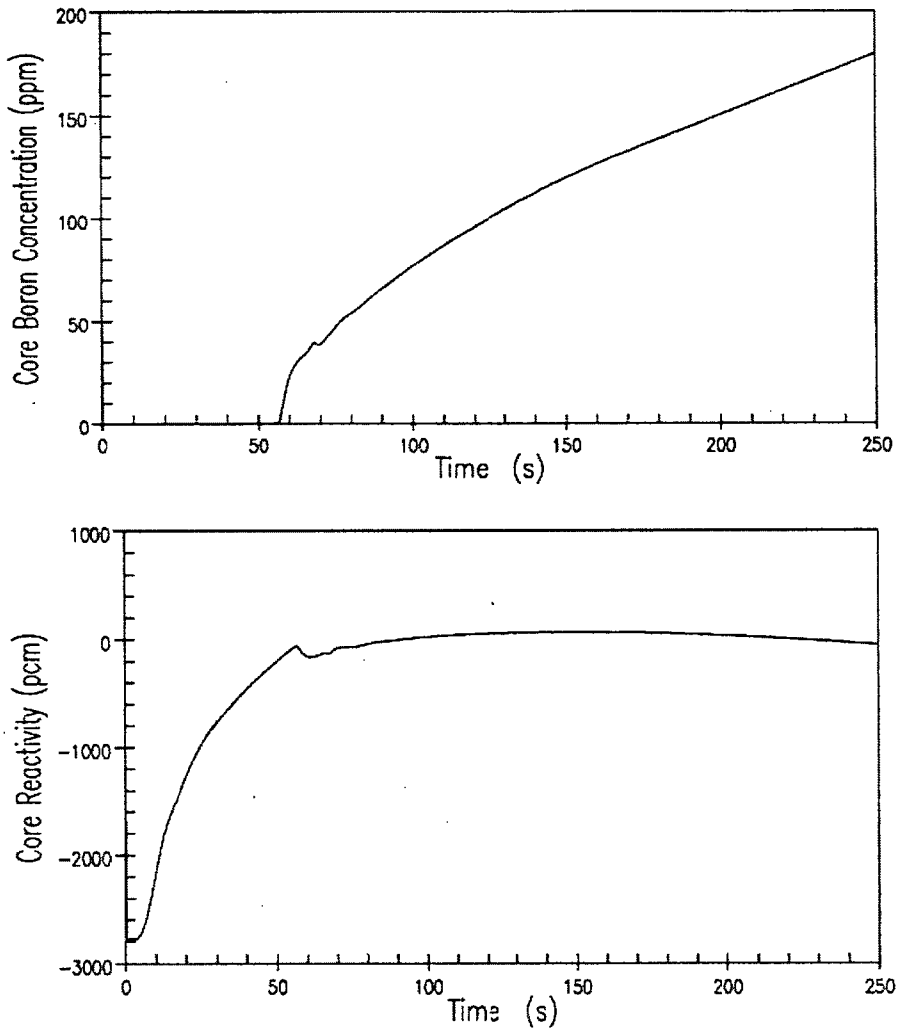
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 1 (two loops in service) With offsite power Figure 14.2.5-1 (sheet 2 of 4)

FIGURE 14.2.5-1
Sheet 3 of 4
RUPTURE OF A STEAM PIPE
UNIT 1 (two loops in service) WITH OFFSITE POWER



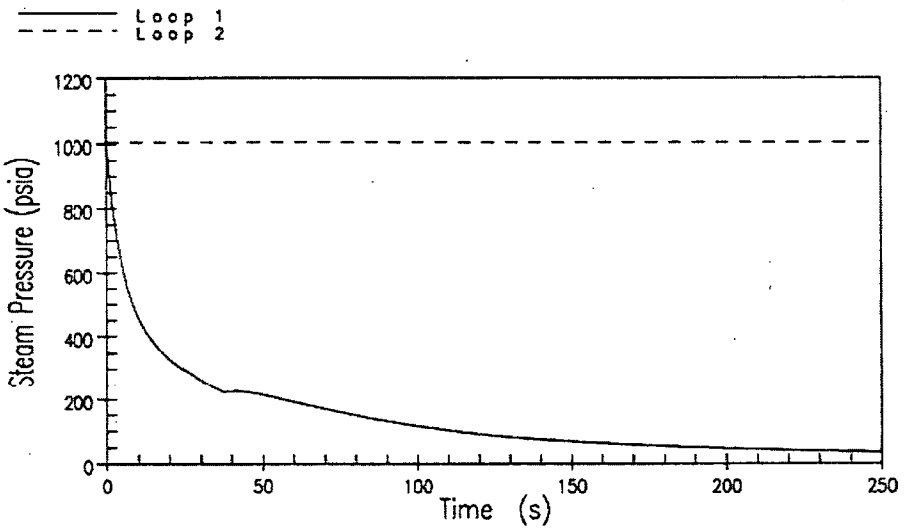
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 1 (two loops in service) With offsite power Figure 14.2.5-1 (sheet 3 of 4)

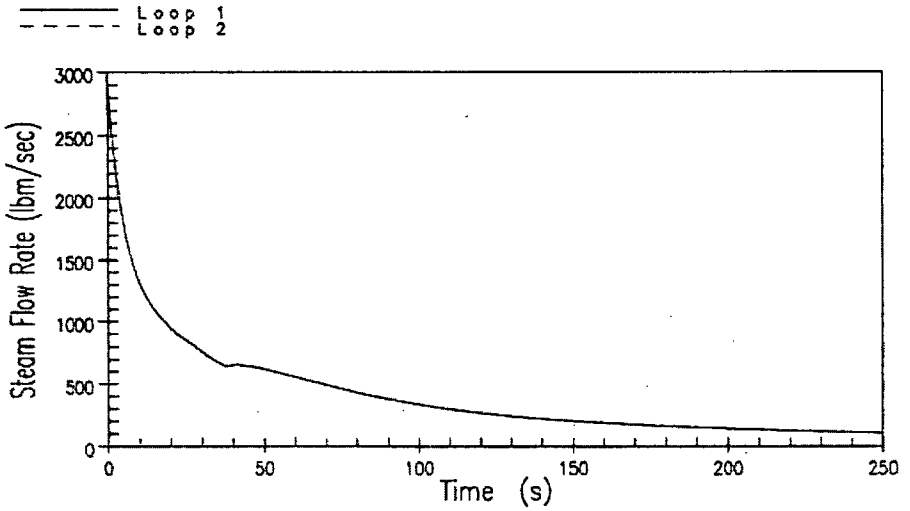
FIGURE 14.2.5-1
Sheet 4 of 4
RUPTURE OF A STEAM PIPE
UNIT 1 (two loops in service) WITH OFFSITE POWER



Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 1 (two loops in service) With offsite power Figure 14.2.5-1 (sheet 4 of 4)

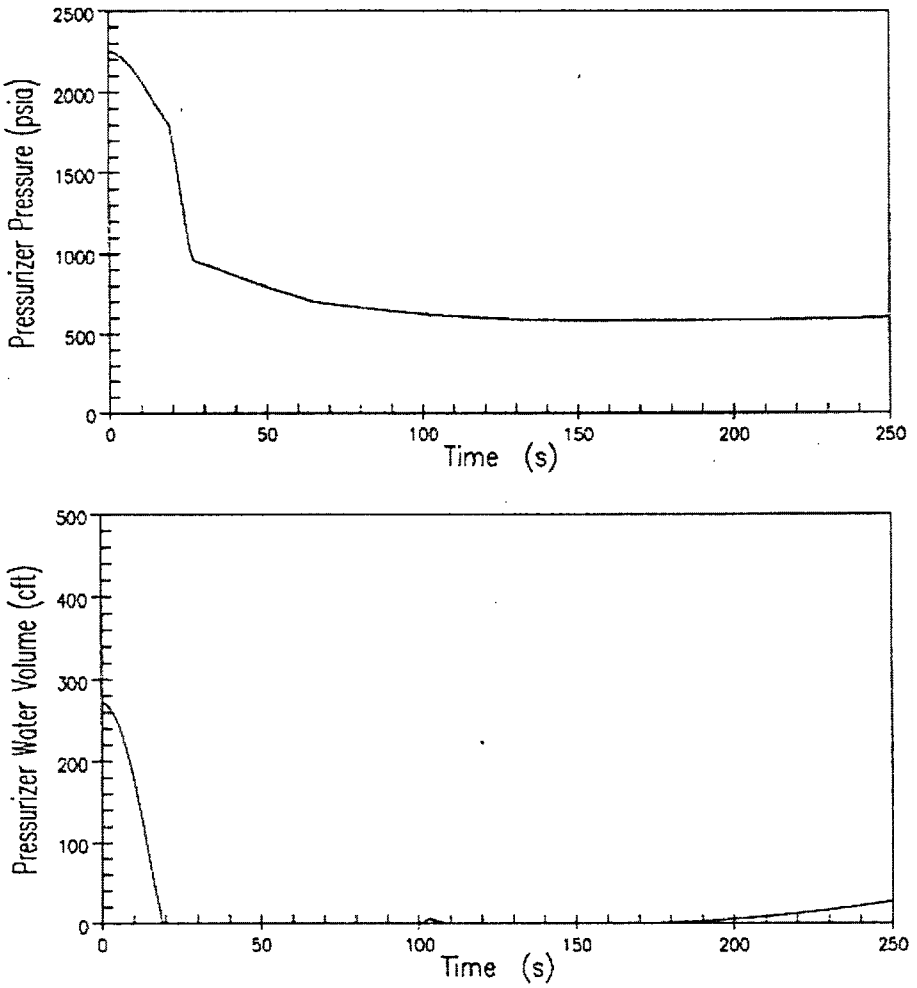
FIGURE 14.2.5-2
Sheet 1 of 4
RUPTURE OF A STEAM PIPE
UNIT 1 (one loop in service) WITH OFFSITE POWER





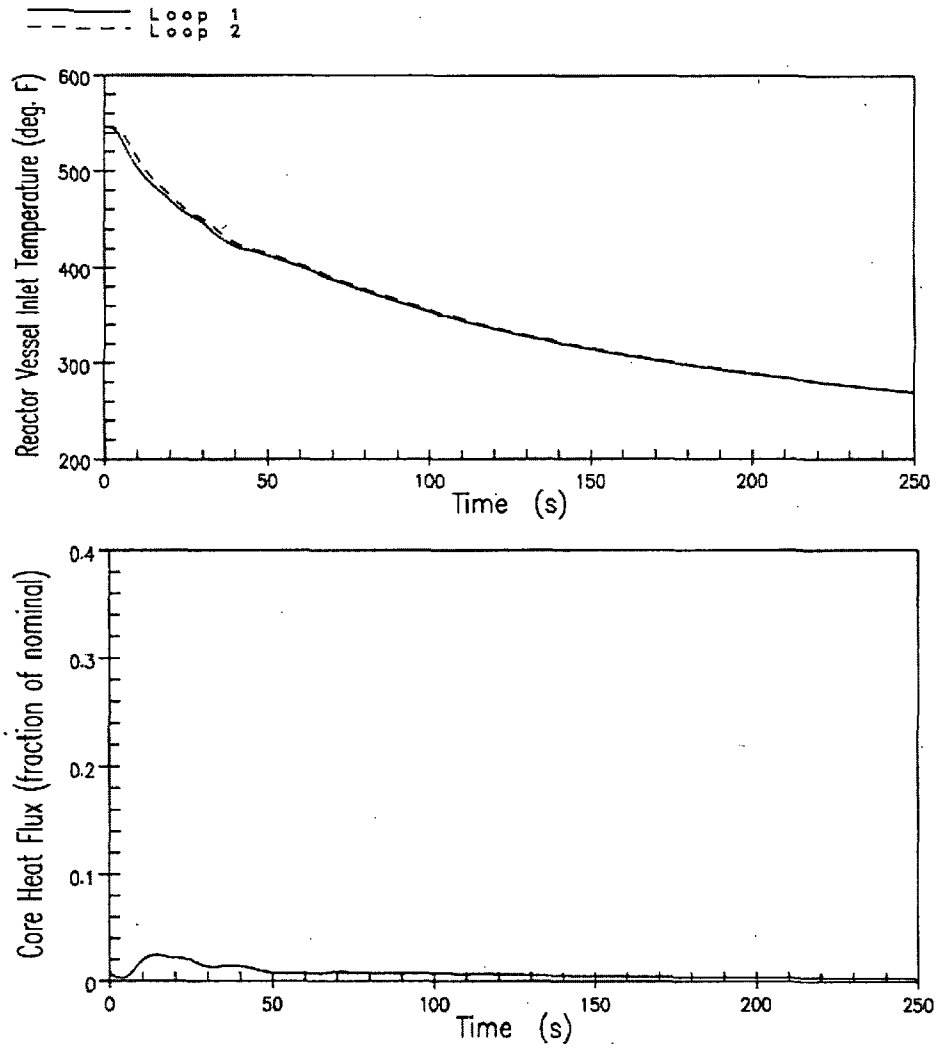
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 1 (one loop in service) With offsite power Figure 14.2.5-2 (sheet 1 of 4)

FIGURE 14.2.5-2
Sheet 2 of 4
RUPTURE OF A STEAM PIPE
UNIT 1 (one loop in service) WITH OFFSITE POWER



Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 1 (one loop in service) With offsite power Figure 14.2.5-2 (sheet 2 of 4)

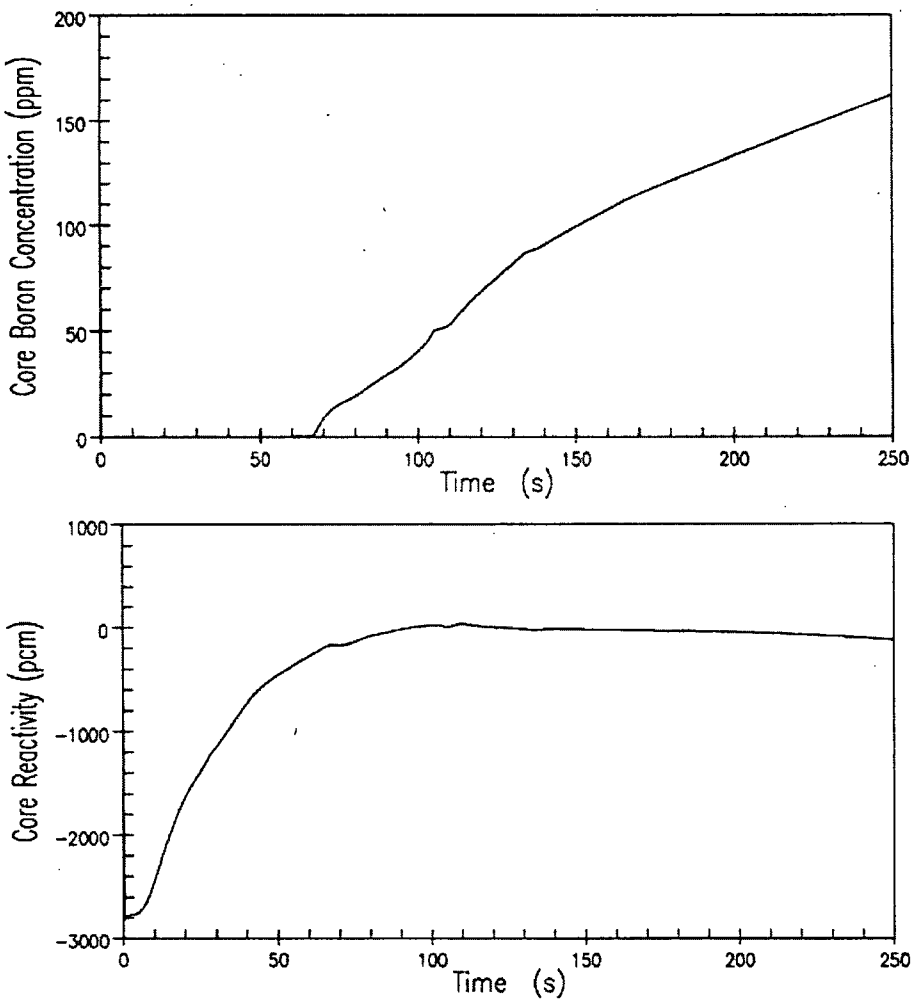
FIGURE 14.2.5-2
Sheet 3 of 4
RUPTURE OF A STEAM PIPE
UNIT 1 (one loop in service) WITH OFFSITE POWER



Point Beach Nuclear Plant
Units 1 and 2

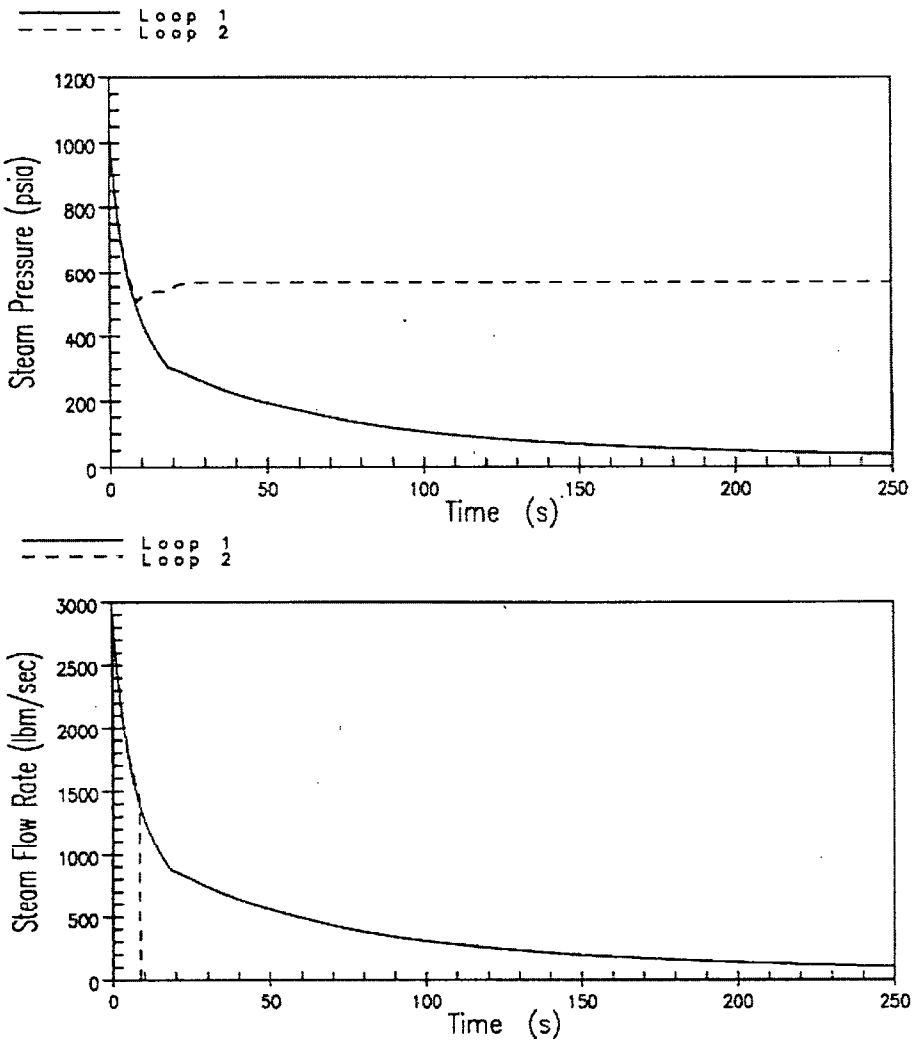
Rupture of a Steam Pipe
Unit 1 (one loop in service)
With offsite power
Figure 14.2.5-2
(sheet 3 of 4)

FIGURE 14.2.5-2
Sheet 4 of 4
RUPTURE OF A STEAM PIPE
UNIT 1 (one loop in service) WITH OFFSITE POWER



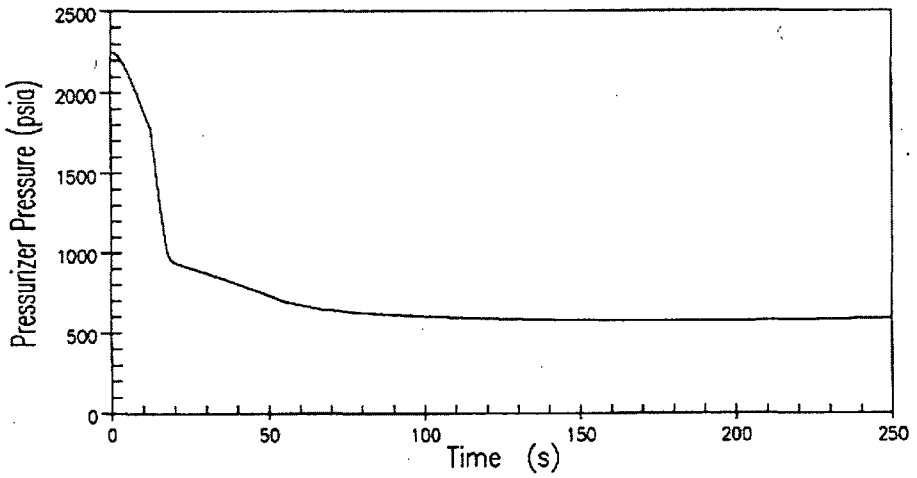
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 1 (one loop in service) With offsite power Figure 14.2.5-2 (sheet 4 of 4)

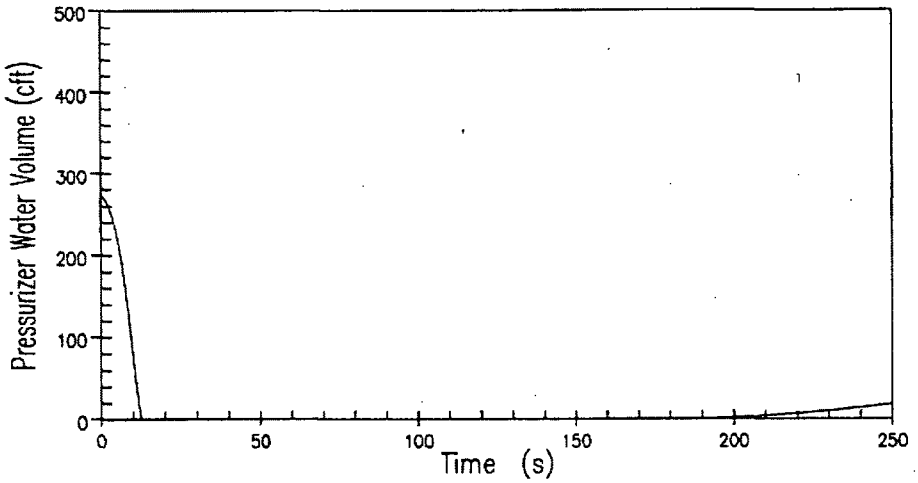
FIGURE 14.2.5-3
Sheet 1 of 4
RUPTURE OF A STEAM PIPE
UNIT 2 (two loops in service) WITH OFFSITE POWER



Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 2 (two loops in service) With offsite power Figure 14.2.5-3 (sheet 1 of 4)

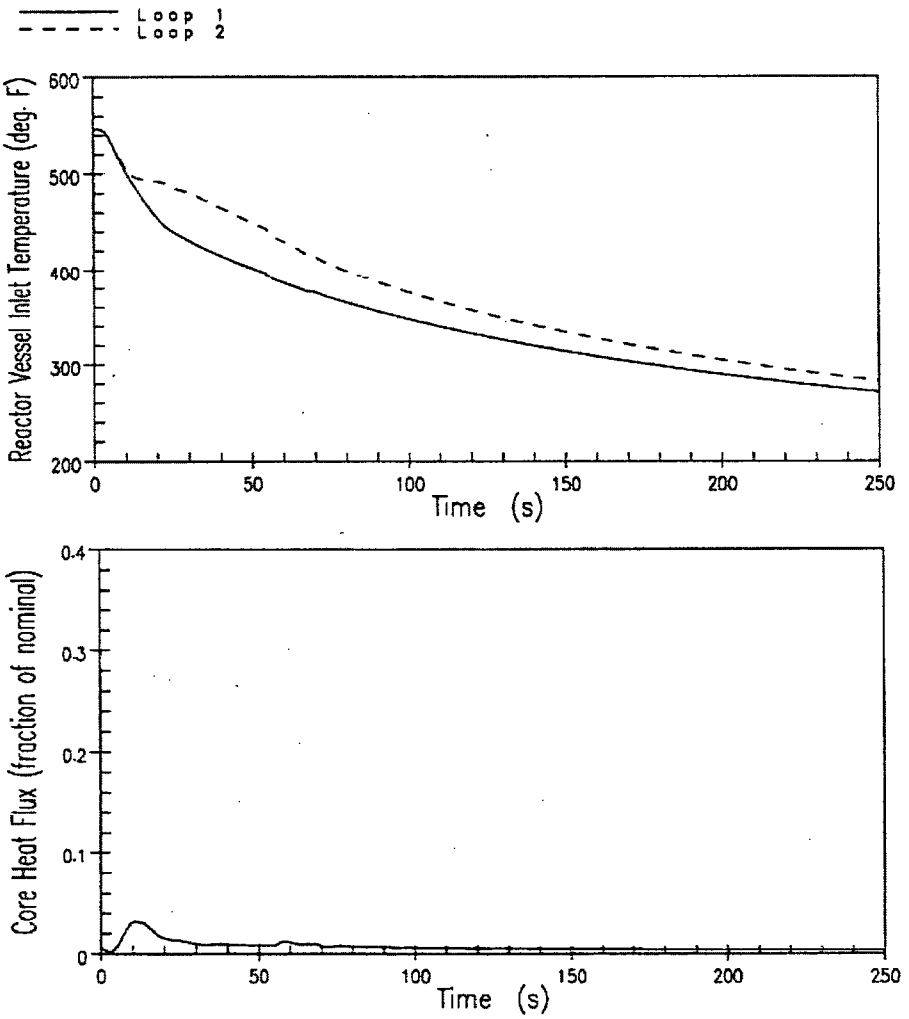
FIGURE 14.2.5-3
Sheet 2 of 4
RUPTURE OF A STEAM PIPE
UNIT 2 (two loops in service) WITH OFFSITE POWER





Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 2 (two loops in service) With offsite power Figure 14.2.5-3 (sheet 2 of 4)

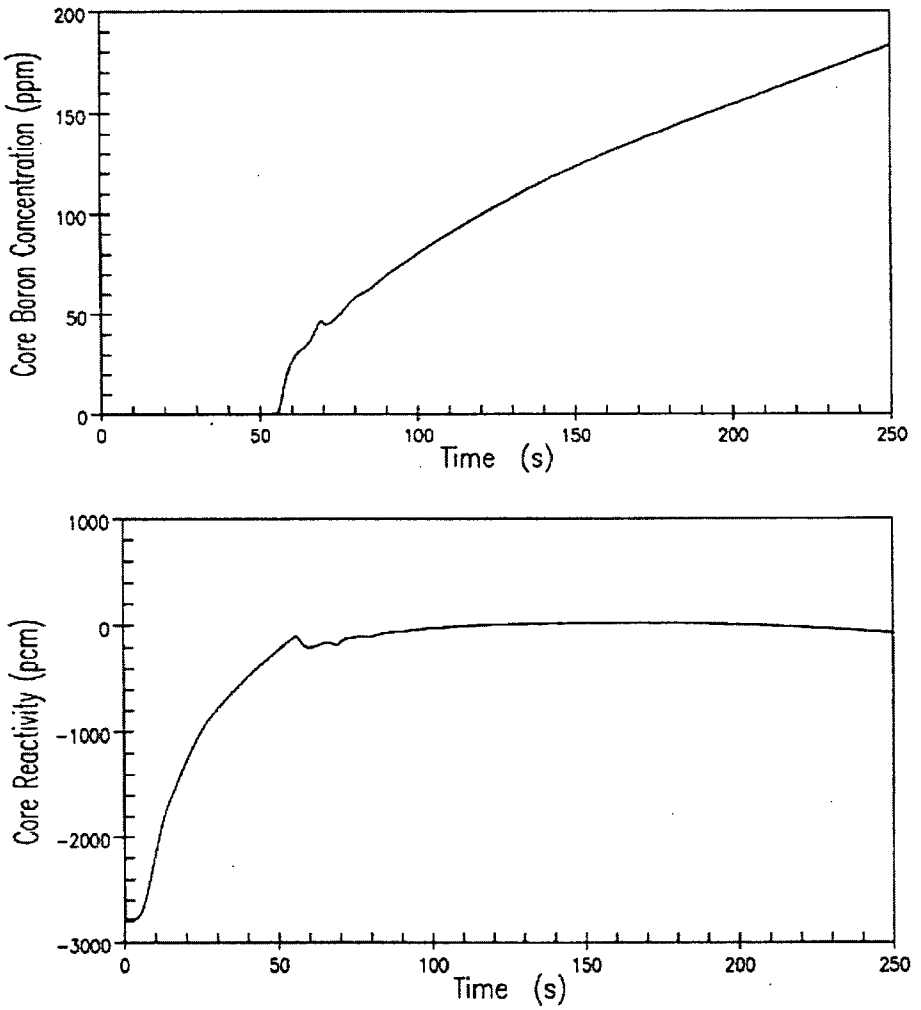
FIGURE 14.2.5-3
Sheet 3 of 4
RUPTURE OF A STEAM PIPE
UNIT 2 (two loops in service) WITH OFFSITE POWER



Point Beach Nuclear Plant
Units 1 and 2

Rupture of a Steam Pipe
Unit 2 (two loops in service)
With offsite power
Figure 14.2.5-3
(sheet 3 of 4)

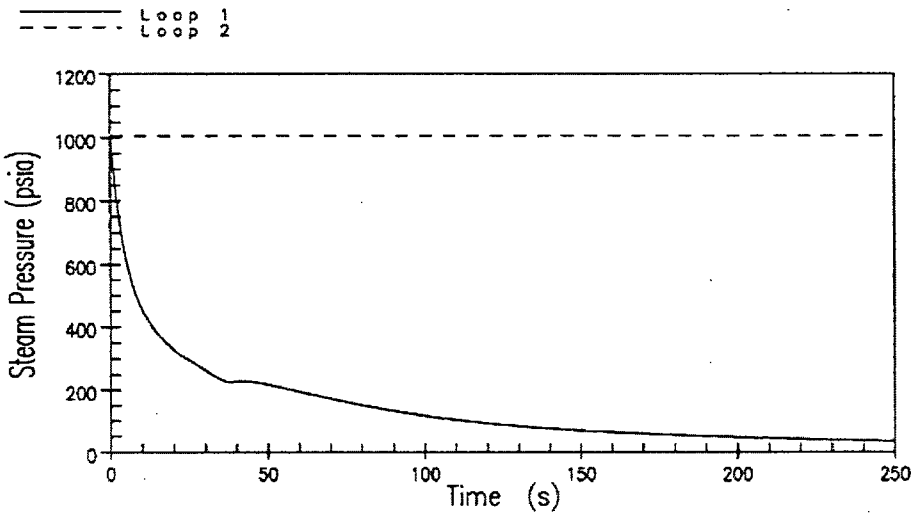
FIGURE 14.2.5-3
Sheet 4 of 4
RUPTURE OF A STEAM PIPE
UNIT 2 (two loops in service) WITH OFFSITE POWER

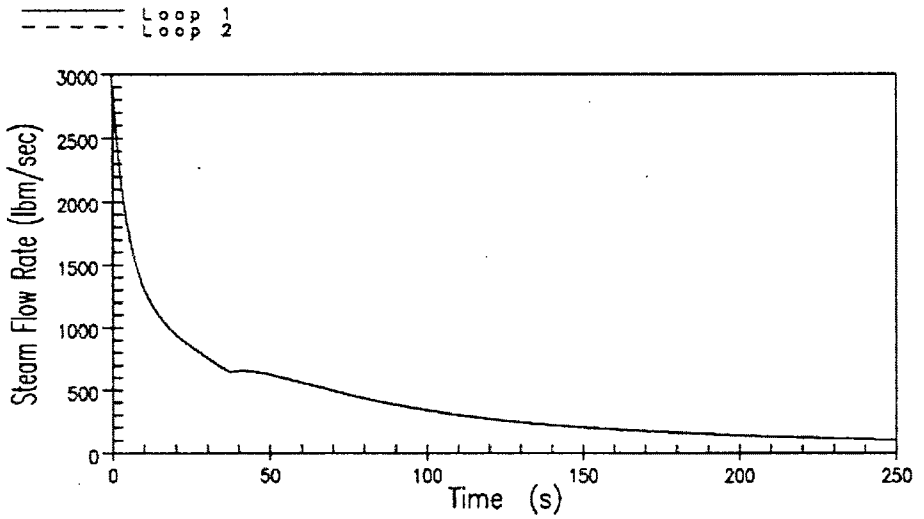


Point Beach Nuclear Plant
Units 1 and 2

Rupture of a Steam Pipe
Unit 2 (two loops in service)
With offsite power
Figure 14.2.5-3
(sheet 4 of 4)

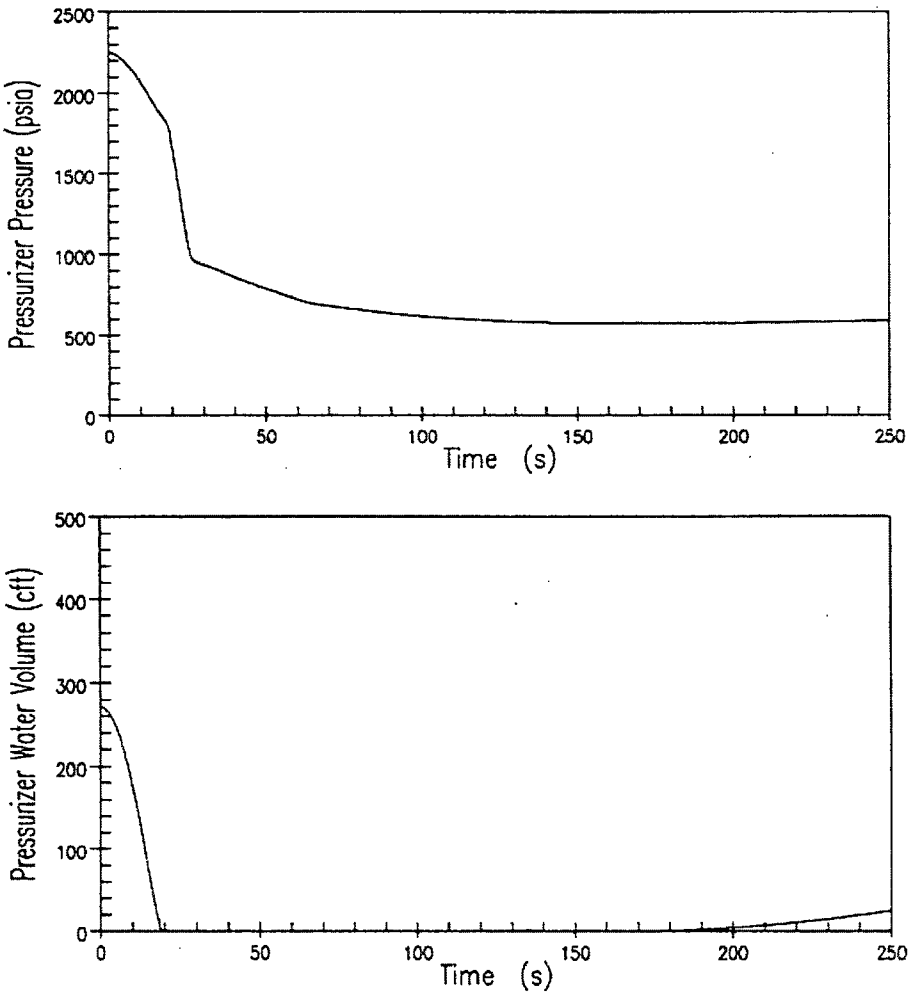
FIGURE 14.2.5-4
Sheet 1 of 4
RUPTURE OF A STEAM PIPE
UNIT 2 (one loop in service) WITH OFFSITE POWER





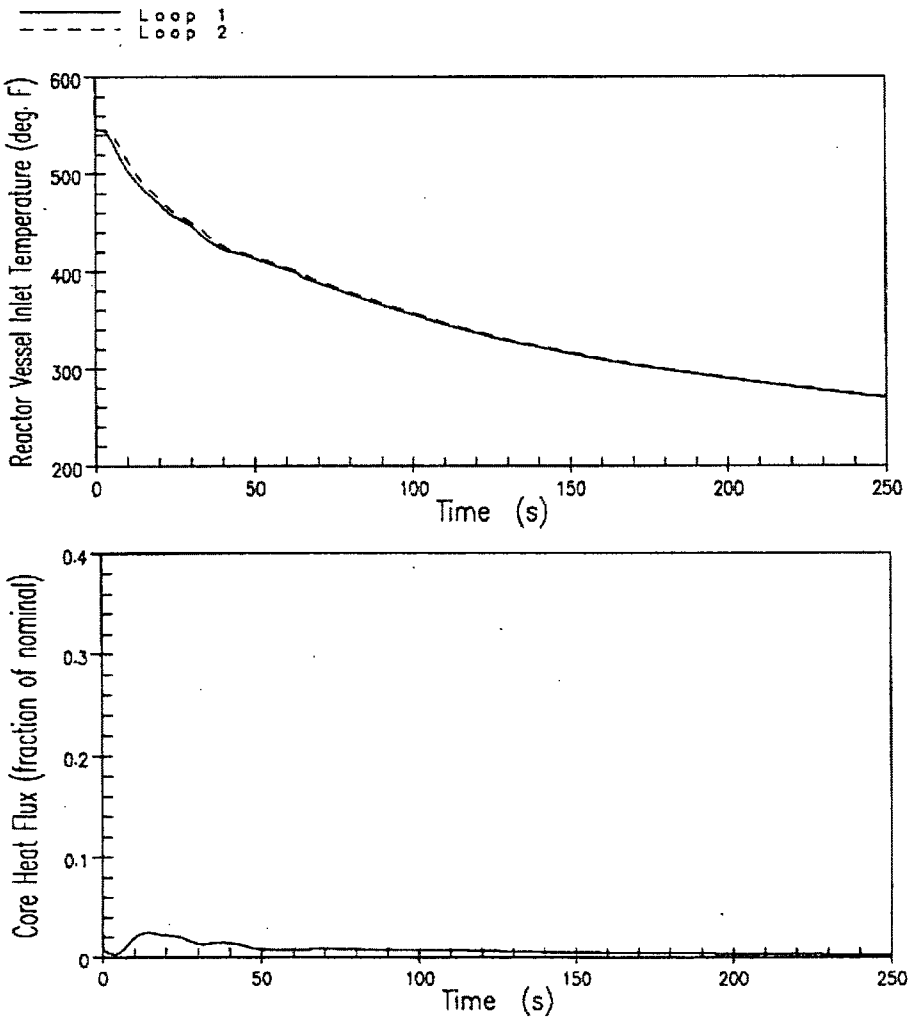
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 2 (one loop in service) With offsite power Figure 14.2.5-4 (sheet 1 of 4)

FIGURE 14.2.5-4
Sheet 2 of 4
RUPTURE OF A STEAM PIPE
UNIT 2 (one loop in service) WITH OFFSITE POWER



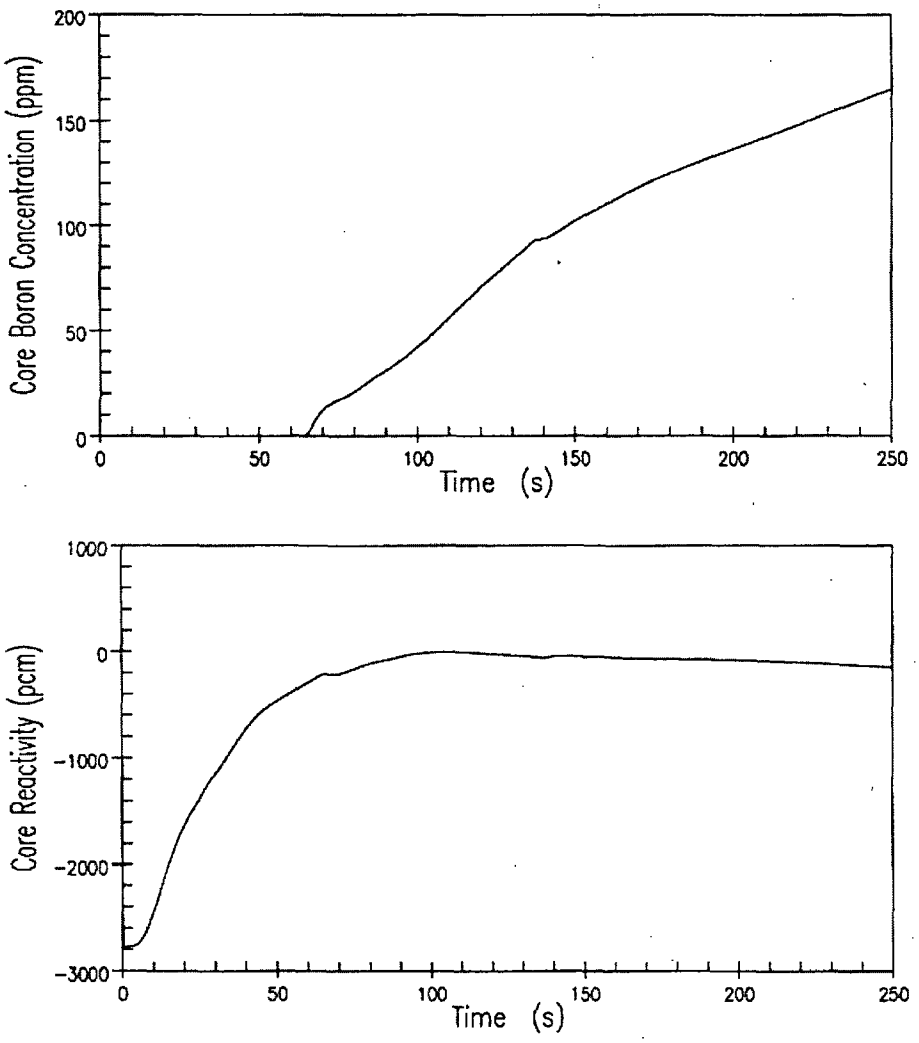
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 2 (one loop in service) With offsite power Figure 14.2.5-4 (sheet 2 of 4)

FIGURE 14.2.5-4
Sheet 3 of 4
RUPTURE OF A STEAM PIPE
UNIT 2 (one loop in service) WITH OFFSITE POWER



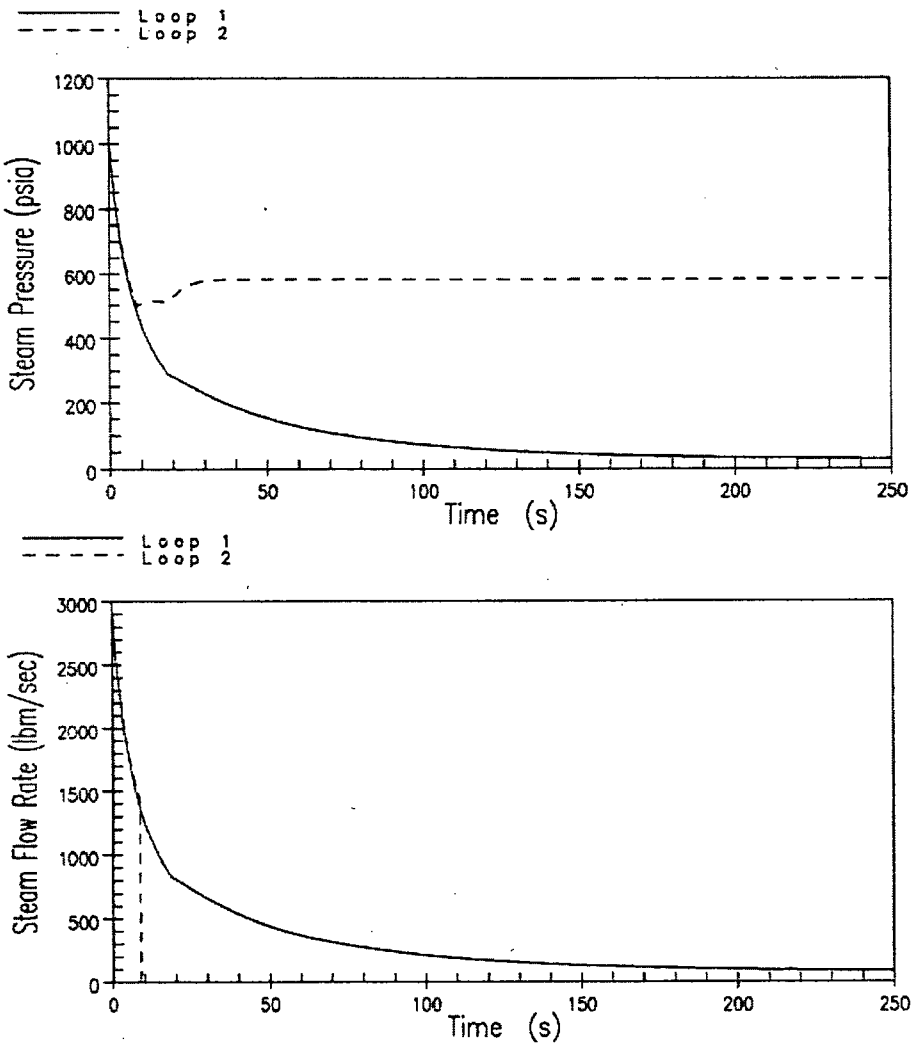
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 2 (one loop in service) With offsite power Figure 14.2.5-4 (sheet 3 of 4)

FIGURE 14.2.5-4
Sheet 4 of 4
RUPTURE OF A STEAM PIPE
UNIT 2 (one loop in service) WITH OFFSITE POWER



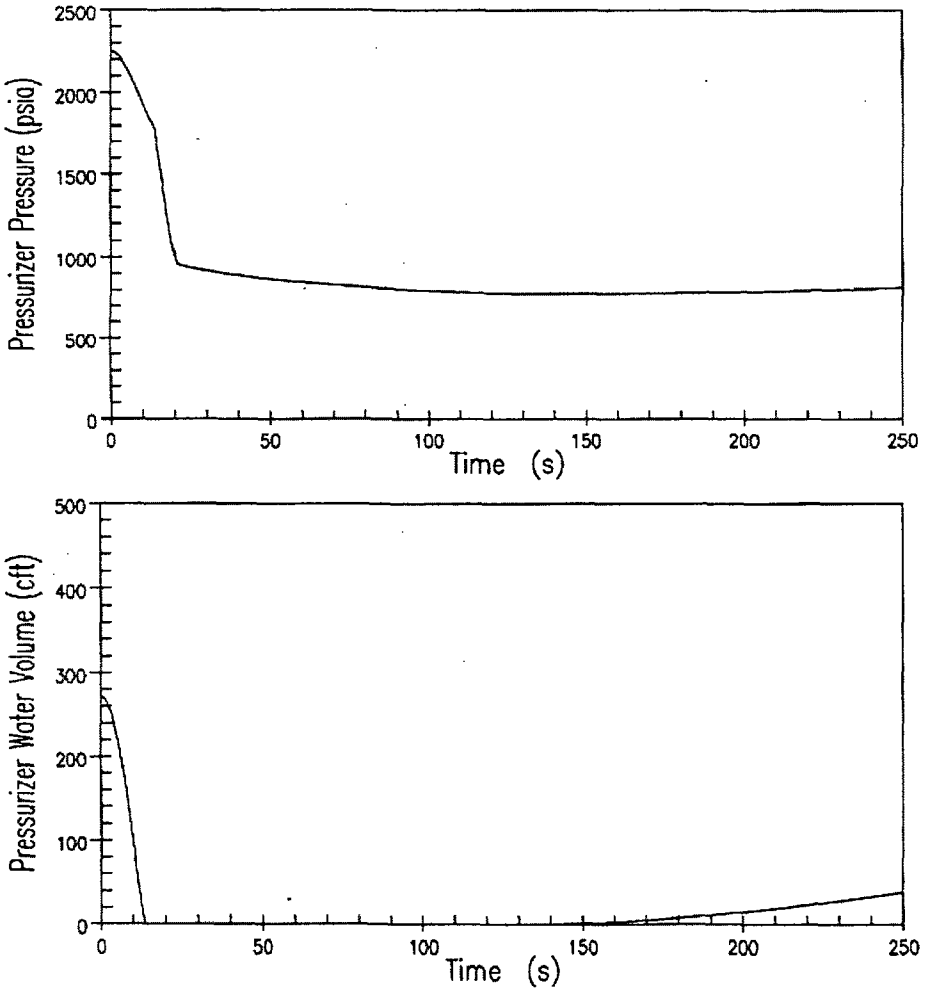
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 2 (one loop in service) With offsite power Figure 14.2.5-4 (sheet 4 of 4)

FIGURE 14.2.5-5
Sheet 1 of 4
RUPTURE OF A STEAM PIPE
UNIT 1 (two loops in service) WITHOUT OFFSITE POWER



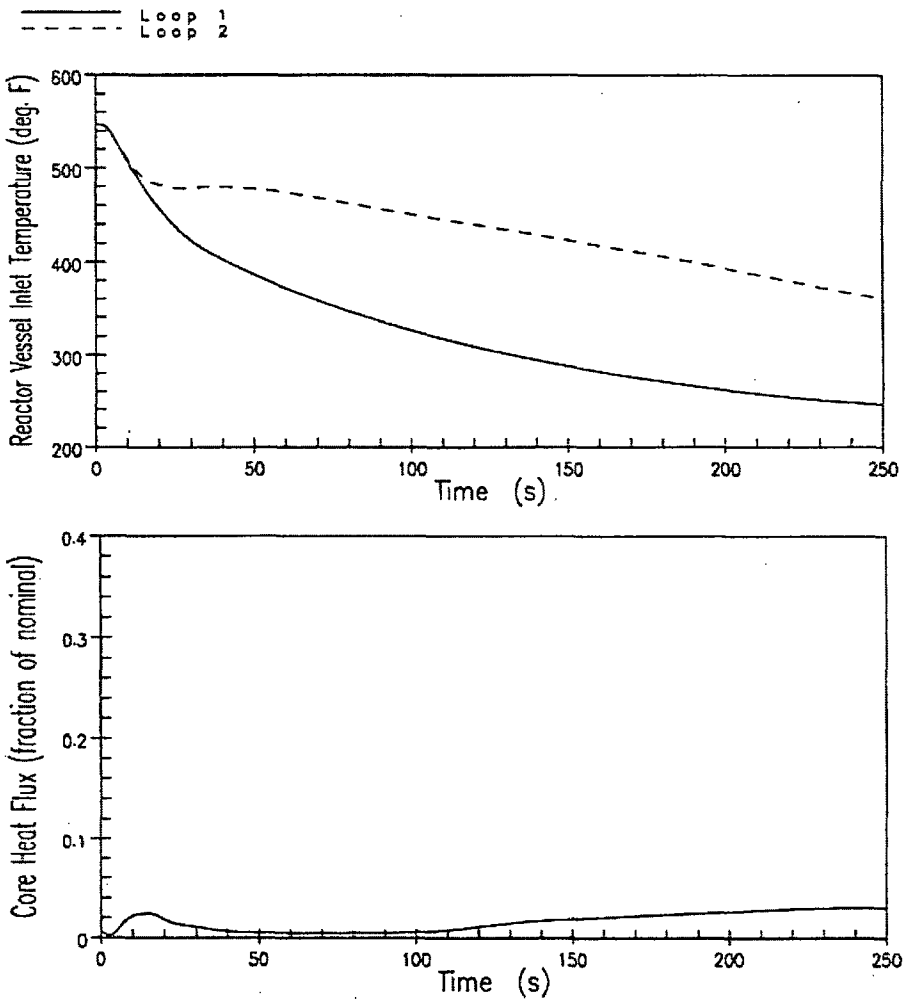
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 1 (two loops in service) Without offsite power Figure 14.2.5-5 (sheet 1 of 4)

FIGURE 14.2.5-5
Sheet 2 of 4
RUPTURE OF A STEAM PIPE
UNIT 1 (two loops in service) WITHOUT OFFSITE POWER



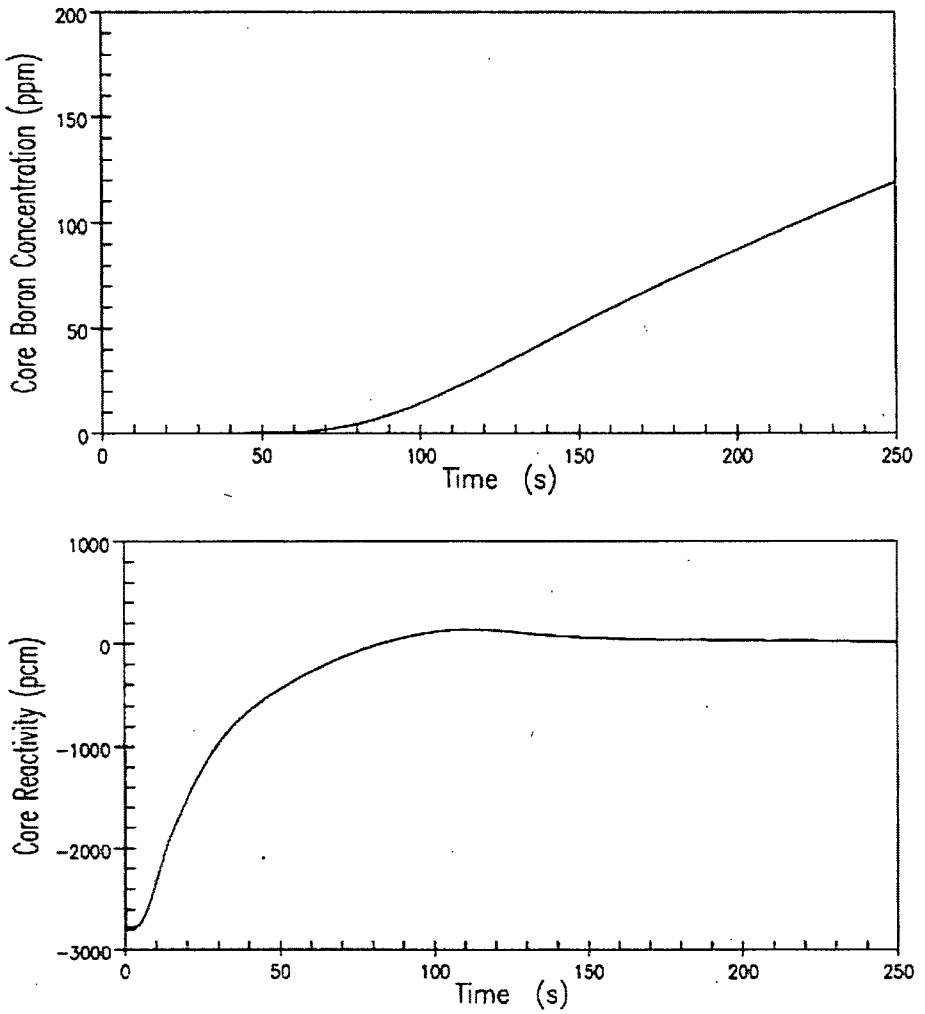
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 1 (two loops in service) Without offsite power Figure 14.2.5-5 (sheet 2 of 4)

FIGURE 14.2.5-5
Sheet 3 of 4
RUPTURE OF A STEAM PIPE
UNIT 1 (two loops in service) WITHOUT OFFSITE POWER



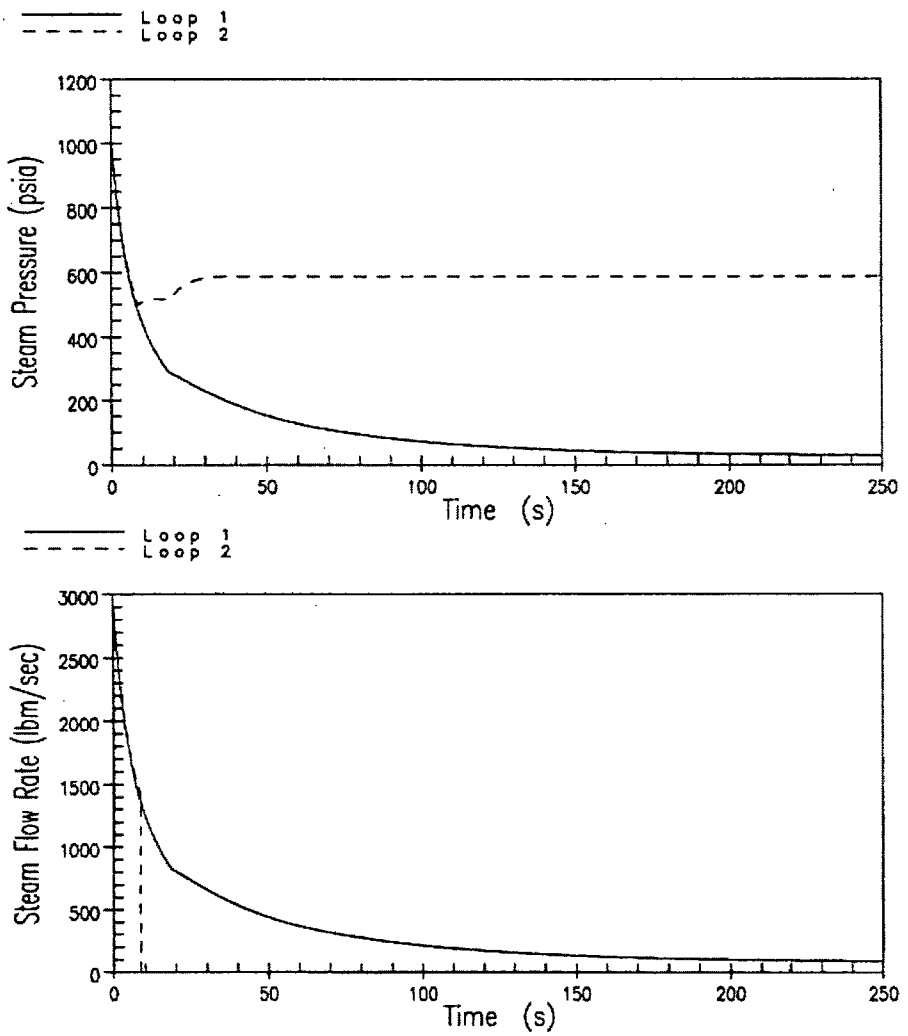
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 1 (two loops in service) Without offsite power Figure 14.2.5-5 (sheet 3 of 4)

FIGURE 14.2.5-5
Sheet 4 of 4
RUPTURE OF A STEAM PIPE
UNIT 1 (two loops in service) WITHOUT OFFSITE POWER



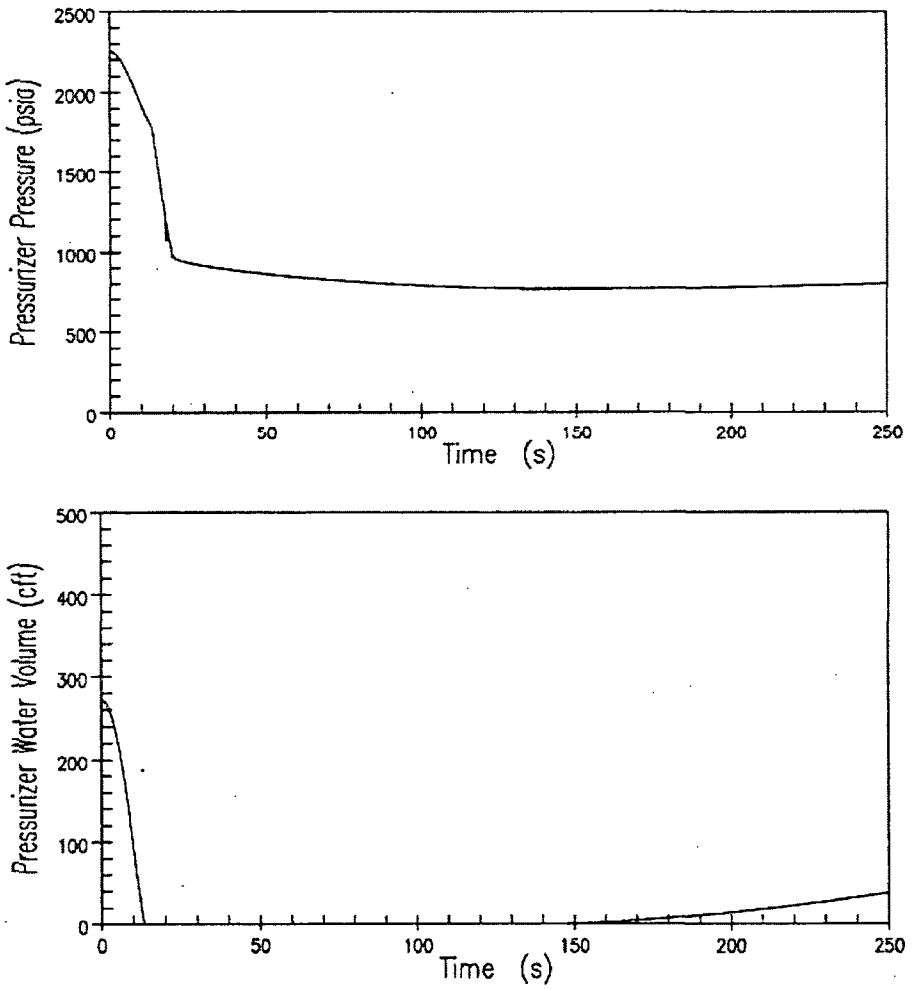
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 1 (two loops in service) Without offsite power Figure 14.2.5-5 (sheet 4 of 4)

FIGURE 14.2.5-6
Sheet 1 of 4
RUPTURE OF A STEAM PIPE
UNIT 2 (two loops in service) WITHOUT OFFSITE POWER



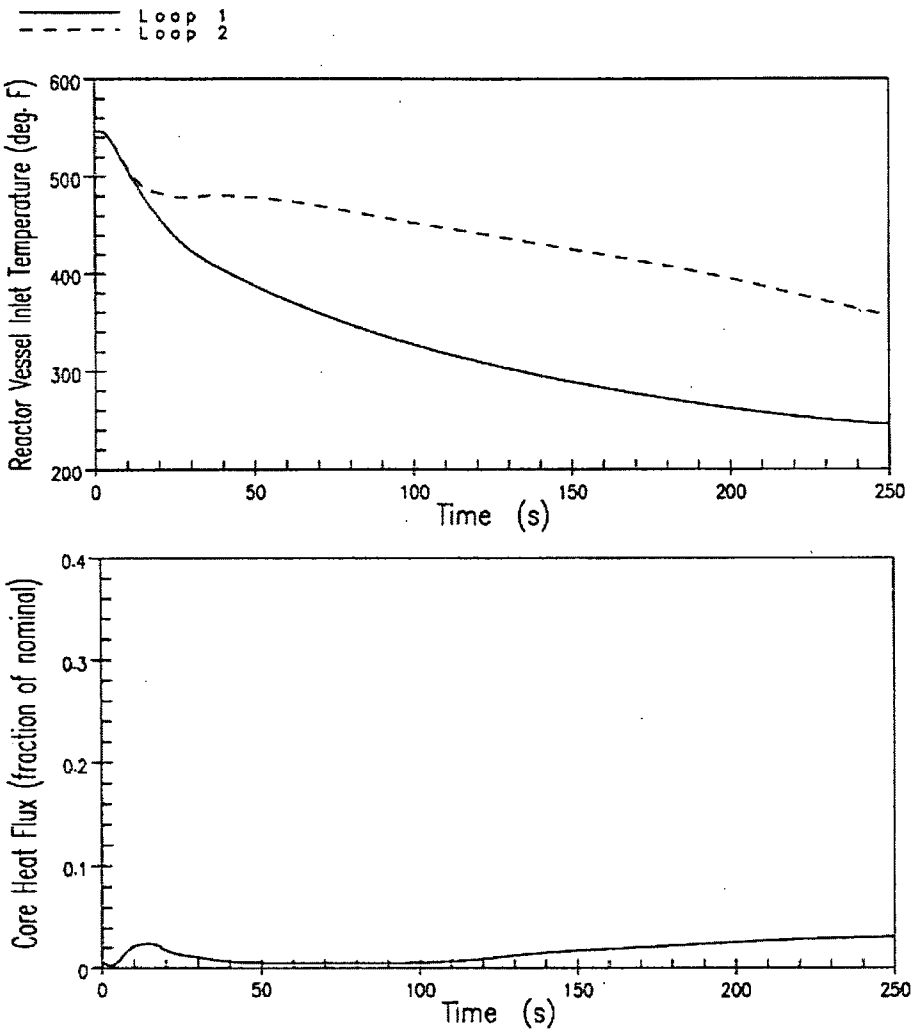
Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 2 (two loops in service) Without offsite power Figure 14.2.5-6 (sheet 1 of 4)

FIGURE 14.2.5-6
Sheet 2 of 4
RUPTURE OF A STEAM PIPE
UNIT 2 (two loops in service) WITHOUT OFFSITE POWER



Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 2 (two loops in service) Without offsite power Figure 14.2.5-6 (sheet 2 of 4)

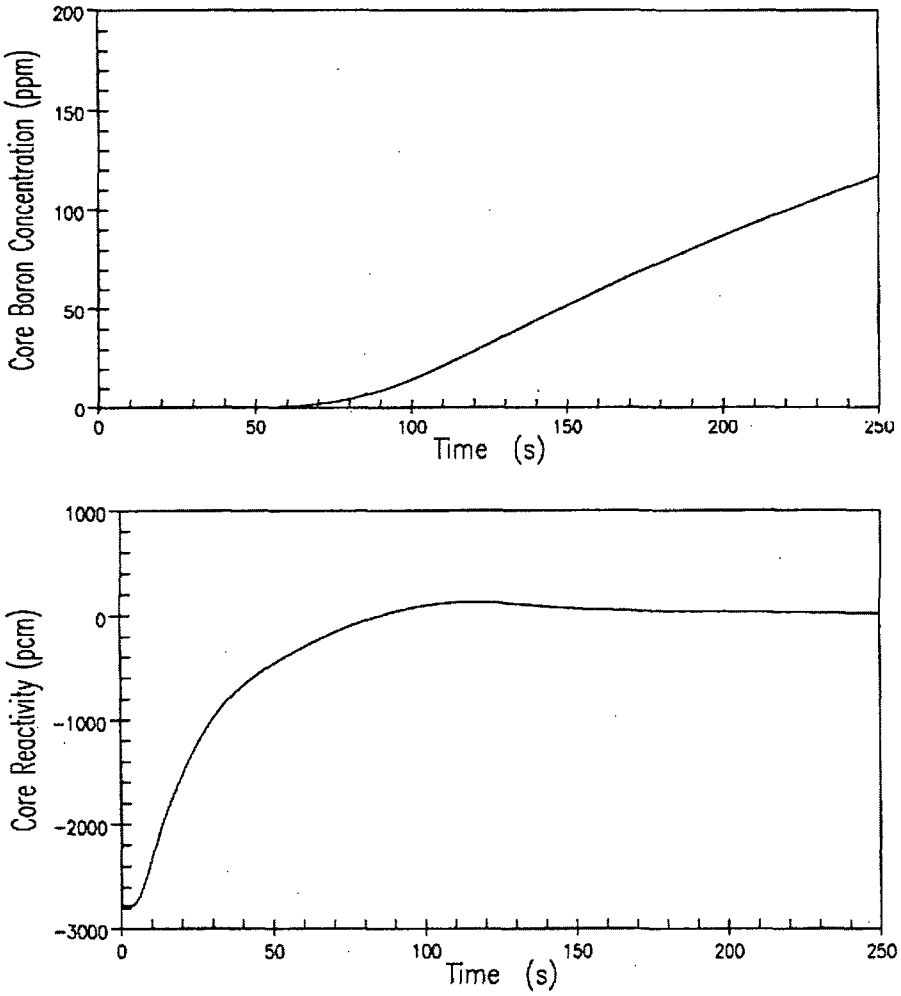
FIGURE 14.2.5-6
Sheet 3 of 4
RUPTURE OF A STEAM PIPE
UNIT 2 (two loops in service) WITHOUT OFFSITE POWER



Point Beach Nuclear Plant
Units 1 and 2

Rupture of a Steam Pipe
Unit 2 (two loops in service)
Without offsite power
Figure 14.2.5-6
(sheet 3 of 4)

FIGURE 14.2.5-6
Sheet 4 of 4
RUPTURE OF A STEAM PIPE
UNIT 2 (two loops in service) WITHOUT OFFSITE POWER



Point Beach Nuclear Plant Units 1 and 2
Rupture of a Steam Pipe Unit 2 (two loops in service) Without offsite power Figure 14.2.5-6 (sheet 4 of 4)

FIGURE 14.2.5-7
CONTAINMENT PRESSURE
MSLB CONTAINMENT RESPONSE ANALYSIS

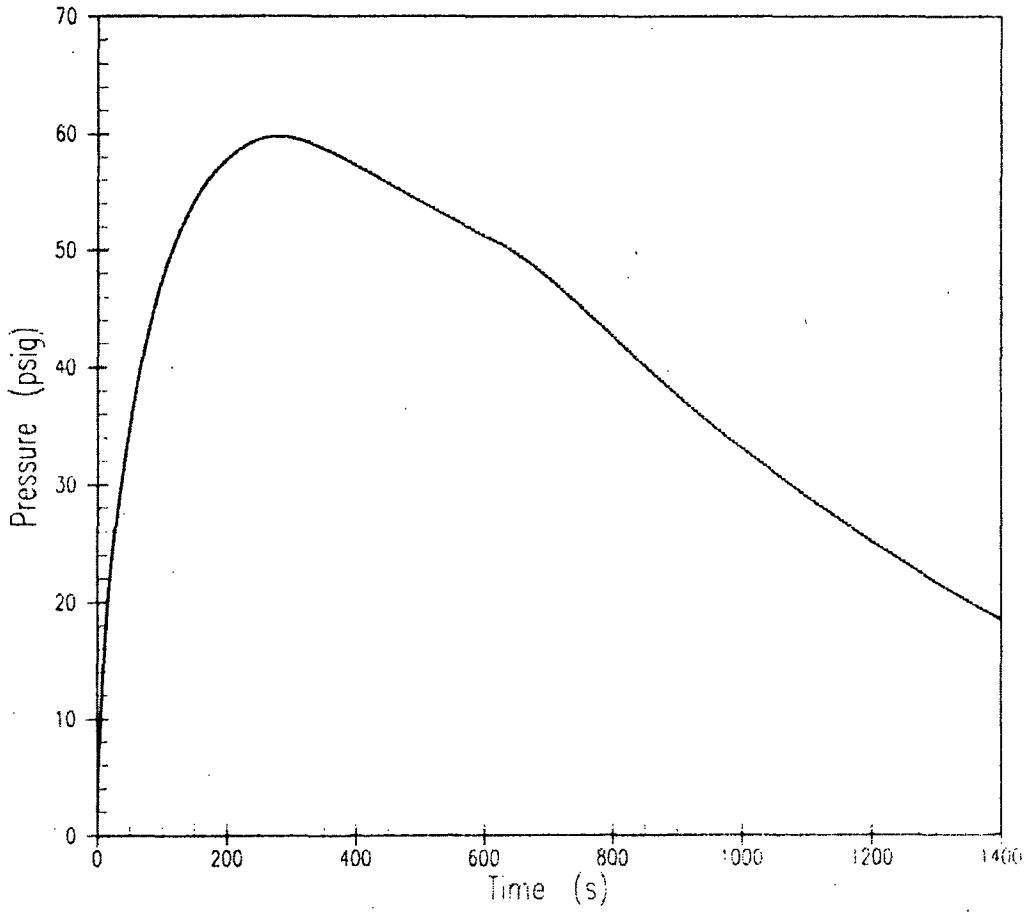
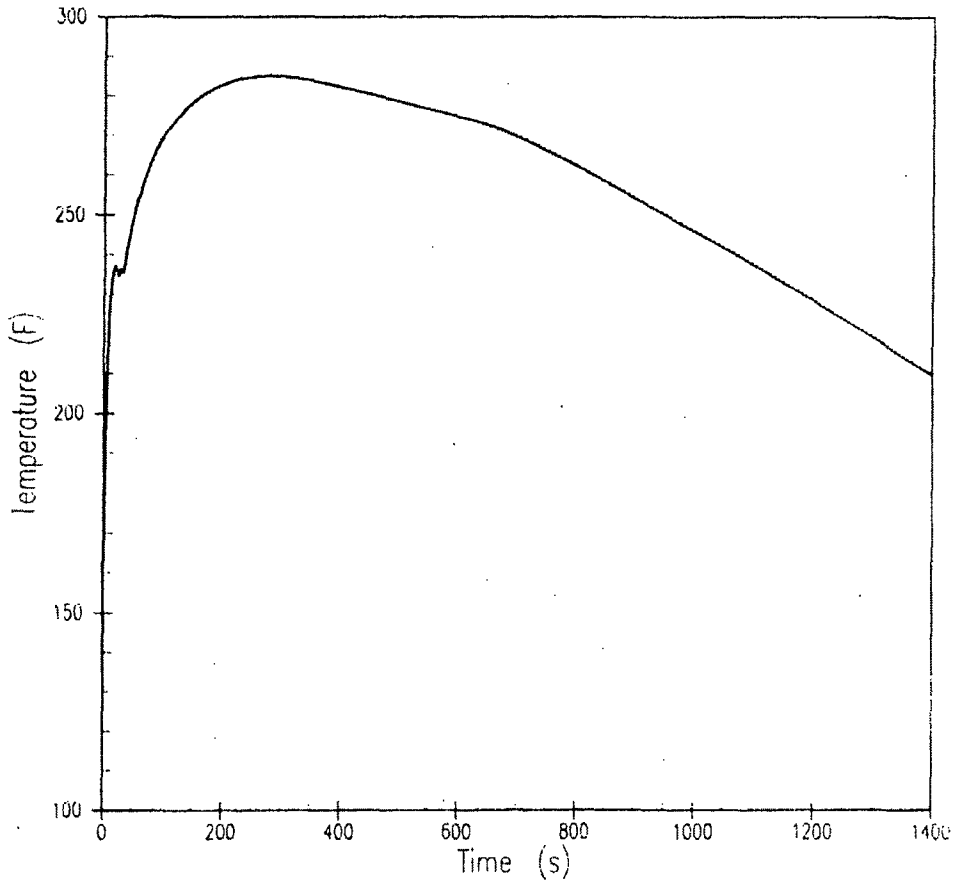


FIGURE 14.2.5-8
CONTAINMENT TEMPERATURE
MSLB CONTAINMENT RESPONSE ANALYSIS



14.2.6 Rupture Of A Control Rod Mechanism Housing - RCCA Ejection

In order for this accident to occur, a rupture of the control rod mechanism housing must be postulated, creating a full system pressure differential acting on the drive shaft. The resultant core thermal power excursion is limited by the Doppler reactivity effects of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals.

A failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

1. Each control rod drive mechanism housing is completely assembled and shop-tested at 3105 psig (nominal).

2. Stress levels in the mechanism are not affected by system transients at power, or by the thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the ASME code, Section III, for Class 1 components.

3. The latch mechanism housing and rod travel housing are Grade F316 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are fabricated with full penetration welds.

Nuclear Design

Even if a rupture of a RCCA drive mechanism housing is postulated, the operation of a plant utilizing chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the RCCAs inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated by boron changes. Further, the location and grouping of control RCCA banks are selected during the nuclear design to lessen the severity of a RCCA ejection accident. Therefore, should a RCCA be ejected from its normal position during full power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCAs is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. Operating instructions require boration at the low-low alarm.

Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference 4. The protection for this accident is provided by high neutron flux trip (high and low setting). These protection functions are described in detail in Section 7.2 of the FSAR.

Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of a RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings. However, even if damage is postulated, it would not be expected to lead to a more severe transient, since RCCAs are inserted in the core in symmetric patterns, and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that RCCA not to fall on receiving a trip signal; however, this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

Limiting Criteria

This event is classified as an ANS Condition IV incident. Due to the extremely low probability of a RCCA ejection accident, some fuel damage could be considered an acceptable consequence.

Comprehensive studies, both of the threshold of fuel failure and of the threshold or significant conversion of the fuel thermal energy to mechanical energy, have been carried out as part of the SPERT project by the Idaho Nuclear Corporation. Extensive tests of UO_2 zirconium clad fuel rods representative of those in pressurized water reactor type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ significantly from the TREAT results, which indicated that this threshold decreases by about 10% with fuel burnup. The cladding failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure (large fuel dispersal, large pressure rise) even for irradiated rods did not occur below 300 cal/gm.

In view of the above experimental results, criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

a. Average fuel pellet enthalpy at the hot spot below 200 cal/gm (360 Btu/lbm) for irradiated fuel. This bounds non-irradiated fuel which has a slightly higher enthalpy limit.

b. Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits.

c. Fuel melting limited to less than the innermost ten percent of the fuel pellet at the hot spot, even if the average fuel pellet enthalpy is below the limits of criterion (a) above.

Method of Analysis

The calculation of the transient is performed in two stages, first an average core calculation and then a hot region calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator density reactivity. Enthalpy and temperature transients in the hot spot are determined by adding a multiple of the average core energy generation to the hotter rods and performing a transient heat-transfer calculation. The asymptotic power distribution calculated without feedback is pessimistically assumed to persist throughout the transient.

Average Core Analysis

The spatial kinetics computer code, TWINKLE (Ref. 4 in Section 14.0), is used for the average core transient analysis. This code solves the two group neutron diffusion theory kinetic equation in one, two or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 8000 spatial points. The computer code includes a detailed multiregion, transient fuel-cladding-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one dimensional axial kinetics code, since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described in the following) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in Section 14.0.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spots before and after ejection are coincident. This is very conservative, since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel-and cladding transient heat transfer computer code, FACTRAN (Ref. 2 in Section 14.0). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A conservative pellet radial power distribution is used within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandburg-Tong (BST) correlation to determine the film boiling coefficient after DNB. The BST correlation is conservatively used assuming zero bulk fluid quality. The DNB ratio is not calculated, instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady-state temperature distribution to agree with the fuel heat transfer design codes. Further description of FACTRAN appears in Section 14.0.

System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant. The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC (Section 3.2) calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in a plant transient computer code. This code calculates the pressure transient taking into account fluid transport in the reactor coolant system and heat transfer to the steam generators. No credit is taken for the pressure reduction caused by the assumed failure of the control rod pressure housing.

Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 14.2.6-1 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using either three dimensional static methods or by a synthesis method employing one dimensional and two dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties, including an allowance for nuclear power peaking due to densification.

Power distributions before and after ejection for a "worst case" can be found in Ref. 4. During plant startup physics testing, ejected rod worths and power distributions are measured in the zero and full power rodded configurations and compared to values used in the analysis. It has been found that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes have been compared and effective weighting factors determined. These weighting factors take the form of multipliers which when applied to single channel feedbacks correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one dimensional (axial) spatial kinetics method is employed, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to three dimensional analysis (Ref. 4).

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning of life and end of life are adjusted in the nuclear core in order to obtain moderator density coefficient curves which are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as a function of power level using a one dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler defect used is given in Table 14.2.6-1. The Doppler weighting factor will increase under accident conditions, as discussed above.

Delayed Neutron Fraction, β_{eff}

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.70% at beginning of life and 0.50% at end of life for the first cycle. The accident is sensitive to β if the ejected rod worth is equal to or greater than β as in zero power transients. In order to allow for reload cycles, pessimistic estimates of β of 0.49% at beginning of cycle and 0.43% at end of cycle were used in the analysis.

Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 14.2.6-1 and includes the effect of one stuck RCCA. These values are reduced by the ejected rod reactivity. The shutdown reactivity has been simulated by dropping a rod of the required worth into the core. The start of rod motion occurs 0.5 second after the high neutron flux trip point is reached. This delay is assumed to consist of 0.2 second for the instrument channel to produce a signal, 0.15 second for the trip breaker to open and 0.15 second for the coil to release the rods. A curve of trip rod insertion versus time is used which assumes that insertion to the dashpot does not occur until 2.2 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over one second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is a particularly important conservatism for hot full power accidents.

Reactor Protection

Reactor protection for a rod ejection is provided by high neutron flux trip (high and low setting). These protection functions are part of the reactor trip system. No single failure of the reactor trip system will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

Results

Cases are presented for both beginning and end of life at zero and full power.

1. Beginning of Cycle, Full Power

   Control bank D is assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor are conservatively calculated to be 400 pcm and 4.2 respectively. The peak hot spot cladding average temperature is 2317°F. The peak hot spot fuel center temperature reaches melting, and is conservatively assumed at 4900°F. However, melting is restricted to less than 10% of the pellet.

2. Beginning of Cycle, Zero Power

   For this condition, control bank D is assumed to be fully inserted and banks B and C are at their insertion limits. The worst ejected rod is located in control bank D and has a worth of 790 pcm and a hot channel factor of 11.0. The peak hot spot cladding average temperature reaches 2664°F, the fuel center temperature is 3880°F.

3. End of Cycle, Full Power

   Control bank D is assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors are conservatively calculated to be 420 pcm and 5.69 respectively. This results in a peak cladding average temperature of 2258°F. The peak hot spot fuel temperature reaches melting conservatively assumed at 4800°F. However, melting is restricted to less than 10% of the pellet.

4. End of Cycle, Zero Power

The ejected rod worth and hot channel factor for this case are obtained assuming control bank D to be fully inserted and banks C and B at their insertion limit. The results are 930 pcm and 18.0, respectively. The peak cladding average and fuel center temperatures are 2967°F and 4083°F respectively.

A summary of the cases presented above is given in Table 14.2.6-1. The nuclear power and hot spot fuel and cladding temperature transients are presented in Figures 14.2.6-1 through 14.2.6-4.

For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. As discussed previously, the reactor will remain subcritical following reactor trip.

The ejection of an RCCA constitutes a break in the reactor coolant system, located in the reactor pressure vessel head. The effects and consequences of loss of coolant accidents are discussed in Section 14.3. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other loss of coolant accident to recover from the event.

Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 15% of the rods entered DNB based on a detailed three-dimensional THINC analysis.

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits. Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the reactor coolant system.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are undermoderated, and bowing will tend to increase the undermoderation at the hot spot. Since the 14 x 14 fuel design is also undermoderated, the same effect would be observed. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that crossflow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

Conclusions

Conservative analyses indicate that the described fuel and cladding limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the reactor coolant system. The analyses have demonstrated that the fission product release, as a result of a number of fuel rods entering DNB, is limited to less than 15% of the fuel rods in the core. The position with regard to fission product release is, therefore much better than for the double ended coolant pipe break, (the maximum hypothetical accident) for which over 70% of the rods are assumed to release fission products.

Radiological Consequences of a Rod Ejection Accident

This section presents an evaluation of the offsite consequences of a control rod ejection (CRE) accident. The specific analyses conducted for the PBNP offsite consequences were generally accepted by NRC (Reference 8). However, the analyses of control room habitability were subsequently identified for further review. Therefore, a quantifiable description of the control room habitability assessment are not provided herein; however, the general results are provided.

The offsite doses from a postulated CRE accident have been evaluated using the analytical methods and assumptions outlined in Regulatory Guide 1.77 (Reference 9). The assumptions include the existence of a pre-accident iodine spike caused by a reactor transient prior to the rod ejection. The spike has raised the RCS I-131 dose equivalent (DE) iodine concentration to the allowed Technical Specification value of 50 μCi/gram. The noble gas activity concentration in the RCS at the time the accident occurs is based on a fuel defect level of 1.0%. The iodine activity concentration of the secondary coolant at the time of the rod ejection accident is assumed to be equivalent to the Technical Specification limit of 1.0 μCi/g of DE I-131.

As a result of the rod ejection, less than 10% of the fuel rods in the core undergo cladding damage. In determining the offsite doses, it is conservatively assumed that 10% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released to the RCS. Ten percent of the total core activity for both iodines and noble gases is assumed to be in the fuel-cladding gap (Reference 9).

A small fraction (i.e., 0.25%) of the fuel in the core is assumed to melt as a result of the CRE accident. One-half of the iodine activity in the melted fuel is released to the RCS, while all of the noble gas activity in the melted fuel is release to the RCS.

Conservatively, all the iodine and noble gas activity (pre-accident and post-accident) is assumed to be in the RCS when determining offsite doses due to the primary to secondary steam generator tube leakage, and all of the iodine and noble gas activity is assumed to be in the containment when determining offsite doses due to containment leakage. However, 50% of the iodine activity released to the containment is assumed to instantaneously plate out on containment surfaces.

The total steam generator tube leakage rate is assumed to be 0.35 gpm (500 gallons/day) per steam generator, or 0.70 gpm (1000 gallons/day) total. Technical Specifications provide a basis for this assumption by the establishment of: 1) a primary-to-secondary operational leakage limit of 150 gallons/day per Steam Generator, and 2) a Steam Generator Program which includes structural integrity and accident induced leakage performance criteria. (Reference 11) Primary-to-secondary system pressure are equalized after 1500 seconds, thus terminating primary-to-secondary leakage in the steam generators. No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. Per Reference 10, an iodine partition factor in the steam generators is assumed to be 0.01 curies/gm of steam per curies/gm of liquid water. 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.

Following the accident, two release paths contribute to the total radiological consequences of the accident. The first is the leakage of radioactivity from the containment atmosphere to the environment and the second is the leakage of radioactivity from the secondary system through the steam generator relief valves. The radioactivity in the containment atmosphere is due to the radioactivity in the primary system coolant that has spilled out of the primary system into the containment through the hole in the reactor head created by the rod ejection. The radioactivity in the secondary system is due to the radioactivity in the primary system coolant that has leaked into the secondary system prior to the accident and also to the radioactivity that is transported to the secondary system by the primary system coolant that leaks through the steam generator tubes during the accident. Steam is released from the steam generator for heat removal purposes because condenser cooling is lost due to the assumed coincident loss of offsite power during the accident.

Core Release Model

The quantity of radioactivity released from the reactor core either to the primary system or to the containment atmosphere during the accident was conservatively calculated using the following assumptions:

1. Ten percent of the fuel rods in the reactor core experience DNB resulting in damage to the clad and all of the noble gases and iodines in the gap of these fuel rods is released. The activity contained in all of the fuel rod gaps consists of 10 percent of the iodines and noble gases accumulated in the reactor core at the end of core life. Less than 15% of the fuel rods in the core enter DNB. The 10% value used in this analysis is supported by WCAP-7588, Revision 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods."

2. One quarter of one percent (0.25 %) of the fuel in the reactor core suffers fuel melt. This value was determined using the following assumptions:

    a. Fifty percent of the fuel rods experiencing clad damage may also experience fuel melting at the centerline of the fuel rod;

    b. Ten percent of the fuel rods that may experience centerline fuel melting actually melt;

    c. Of those fuel rods actually melting, fifty percent of the axial length of the fuel rod melts due to the power distribution;

Containment Release Pathway

The model for this release pathway assumes that all of the radioactivity initially present in the primary system due to the fuel defects and the pre-accident iodine spike and the radioactivity introduced by the fuel rod cladding failures and the melted fuel is instantaneously and homogeneously mixed throughout the net free volume of the containment atmosphere at the time of the accident. Of the radioiodines released to the containment atmosphere, fifty percent instantaneously plate out on the equipment and the structural members within the containment building. No credit is assumed for the removal of the radioiodine from the containment atmosphere by the containment spray system. The only removal processes considered are radioactive decay and leakage.

The containment leak rate for the first 24 hours is its design leak rate of 0.4 percent per day. Thereafter, the containment leak rate is 0.2 percent per day. The iodine and noble gas activity released during the 0 to 30 days post-accident time period is calculated to cause the offsite doses listed in Table 1.

Other assumptions for this dose analysis are presented in Tables 14.1.8-3 and 14.1.8-4. Table 14.1.8-3 provides dose conversion factors, breathing rates, and atmospheric dispersion factors. Table 14.1.8-4 provides the core and coolant activities assumed in the analysis.

Acceptance Criteria

The offsite dose limits for the CRE accident are well within the 10 CFR 100 guideline values (75 rem thyroid and 6 rem whole body dose). The control room dose limits of 30 rem thyroid, 5 rem whole body, and 30 rem beta skin are defined by the Standard Review Plan (Section 6.4).

Results

The calculated offsite dose consequences of the postulated CRE accident are described by Table 14.2.6-3. The calculated control room dose consequences are not presented herein, but are within 10 CFR 100 guideline values. To reduce control room thyroid dose to within the 30 rem limit, credit was taken for the administration of potassium iodide tablets.

Conclusions from CRE Radiological Analyses

The offsite thyroid and whole body doses and the control room whole body and beta skin doses are within the current NRC acceptance criteria for a control rod ejection accident.

References

1. Tong, "Post DNB Heat Transfer," WCAP 7247

2. Redfield, J.A., "CHIC-KIN -- A Fortran Program for Intermediate and Fast Transients in a Water Moderated Reactor," WAPD-TM-479, January, 1965

3. Barry, R. F., "The Revised LEOPARD Code - A Spectrum Dependent Non Spatial Depletion Program," WCAP-2759 (1965)

4. "Power Distribution Control of Westinghouse PWR" WCAP 7208 (1968)

5. Conway and Hein, Journal of Nuclear Materials (15.1), 1965

6. Ogard & Leary, "High Temperature Heat Content and Heat Capacity of Uranium Dioxide - Plutonium Dioxide Solid Solutions," LA-DC-8620

7. WE letter to NRC, NPL-97-0144, "Supplement to Technical Specifications Change Requests 188 and 189," dated April 2, 1997

8. NRC Safety Evaluation Report (SER) dated July 1, 1997, "Issuance of Amendments for Technical Specification Change Requests 188 and 189."

9. US AEC Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974

10. NUREG-0800, Standard Review Plan 15.6.3, "Radiological Consequences of a Steam Generator Tube Rupture (PWR)," Rev. 2, July 1981

11. Amendment Nos. 223 and 229 to Renewed Facility Operating License Nos. DPR-24 and DPR-27 respectively, for the Point Beach Nuclear Plant, Units 1 and 2, respectively, dated August 22, 2006.

TABLE 14.2.6-1
PARAMETERS USED IN THE ANALYSIS OF THE
ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Parameters	BOL-HZP	BOL-HFP	EOL-HZP	EOL-HFP
Initial core power level, percent of 1650.0 MWt	0%	102%	0%	102%
Ejected rod worth, pcm	790	400	930	420
Delayed neutron fraction	0.0049	0.0049	0.0043	0.0043
Doppler reactivity defect (absolute value), pcm	1000	1000	900	900
Doppler feedback reactivity weighting	2.071	1.2	2.704	1.3
Trip reactivity, percent ΔK	2.0	4.0	2.0	4.0
F_q before rod ejection	N/A	2.6	N/A	2.6
F_q after rod ejection	11.0	4.2	18.0	5.69
Number of operational pumps	1	2	1	2
Maximum fuel pellet average temperature, °F	3461	4108	3740	4016
Maximum fuel center temperature, °F	3880	>4900	4083	>4800
Maximum cladding average temperature, °F	2664	2317	2967	2258
Maximum fuel stored energy, cal/gm	146.6	180.1	160.9	175.2
Maximum fuel melt	nil	7.6	nil	8.2

TABLE 14.2.6-2
ASSUMPTIONS USED FOR
CONTROL ROD EJECTION ACCIDENT ANALYSIS

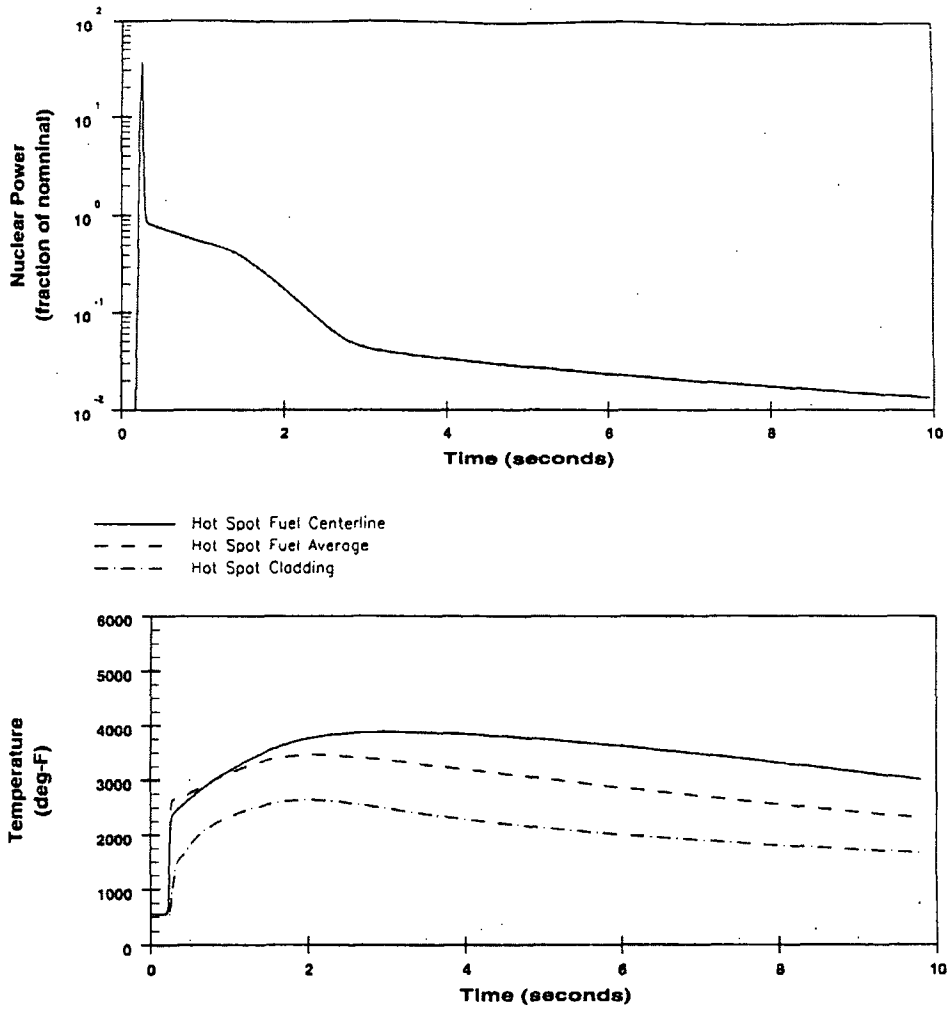
Parameter	Value
Power	1650 MWt
Reactor Coolant Noble Gas Activity (Pre-Accident)	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity (Pre-Accident)	50 μCi/gm of DE I-131
Fraction of Core Activity in Gap	0.10
Iodine Removal in Containment	
Instantaneous Iodine Plateout	50%
Secondary Coolant Activity Prior to Accident	1.0 μCi/gm of DE I-131
Total SG Tube Leak Rate During Accident	0.35 gpm per SG
Iodine Partition Factor in SGs	0.01
Containment Free Volume	1.065×10^6 ft^3
Containment Leak Rate	
0 - 24 hr	0.4% / day
>24 hr	0.2% / day
Steam Release From SGs	
Initial	158,200 lb
Cooldown	428,000 lb

Reference 7.

TABLE 14.2.6-3
OFFSITE DOSES DUE TO THE RADIOACTIVITY RELEASED DURING THE CONTROL ROD EJECTION ACCIDENT (REM)

	Thyroid	Whole Body
Site Boundary (0 - 2 hr)	22	0.2
Low Population Zone (0 - 30 days)	10	0.03

FIGURE 14.2.6-1
RCCA EJECTION TRANSIENT
BEGINNING OF LIFE
ZERO POWER

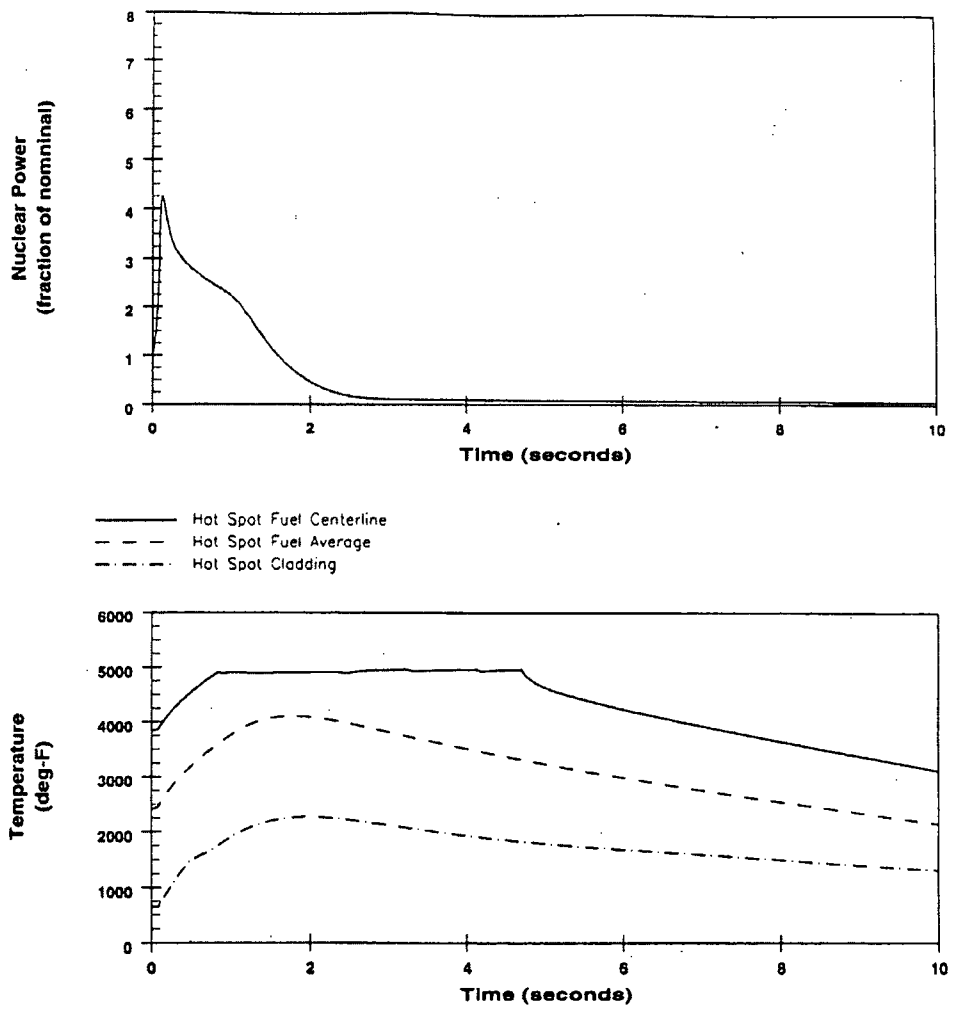


Point Beach Nuclear Plant
Units 1 and 2

RCCA Ejection Transient
Beginning of Life
Zero Power

Figure 14.2.6-1

FIGURE 14.2.6-2
RCCA EJECTION TRANSIENT
BEGINNING OF LIFE
FULL POWER

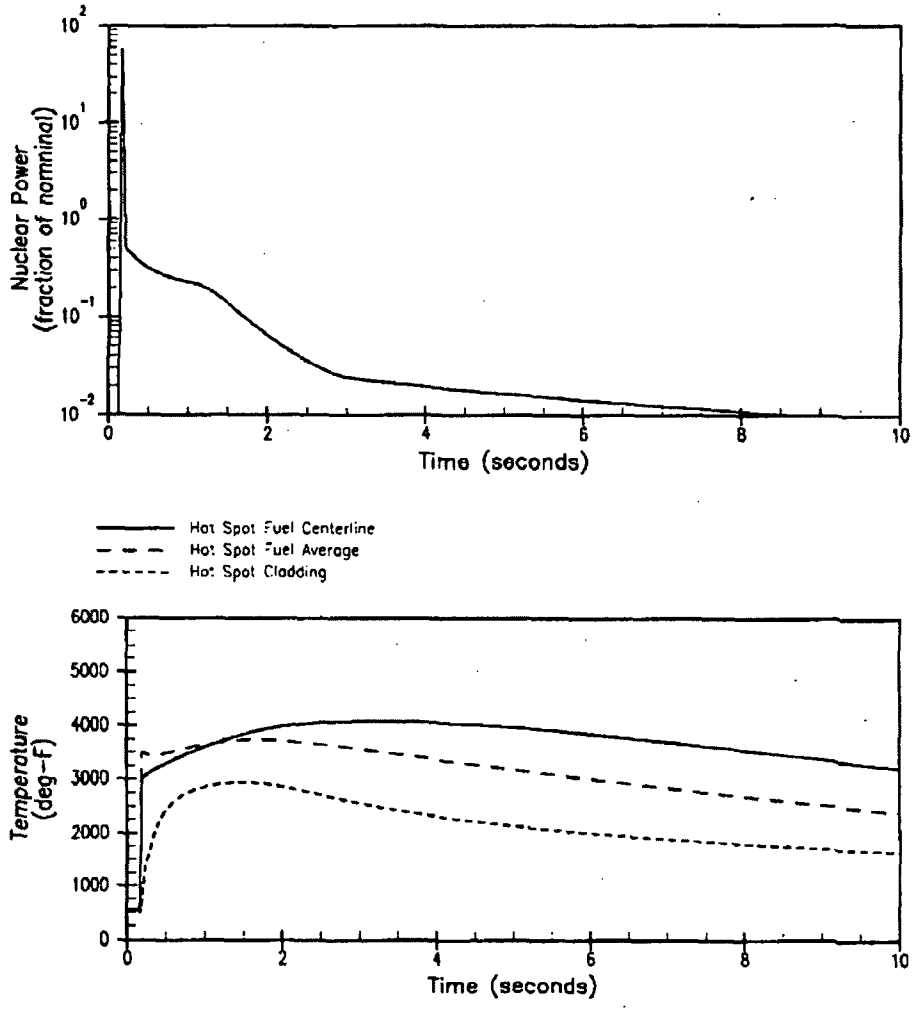


Point Beach Nuclear Plant
Units 1 and 2

RCCA Ejection Transient
Beginning of Life
Full Power

Figure 14.2.6-2

FIGURE 14.2.6-3
RCCA EJECTION TRANSIENT
END OF LIFE
ZERO POWER

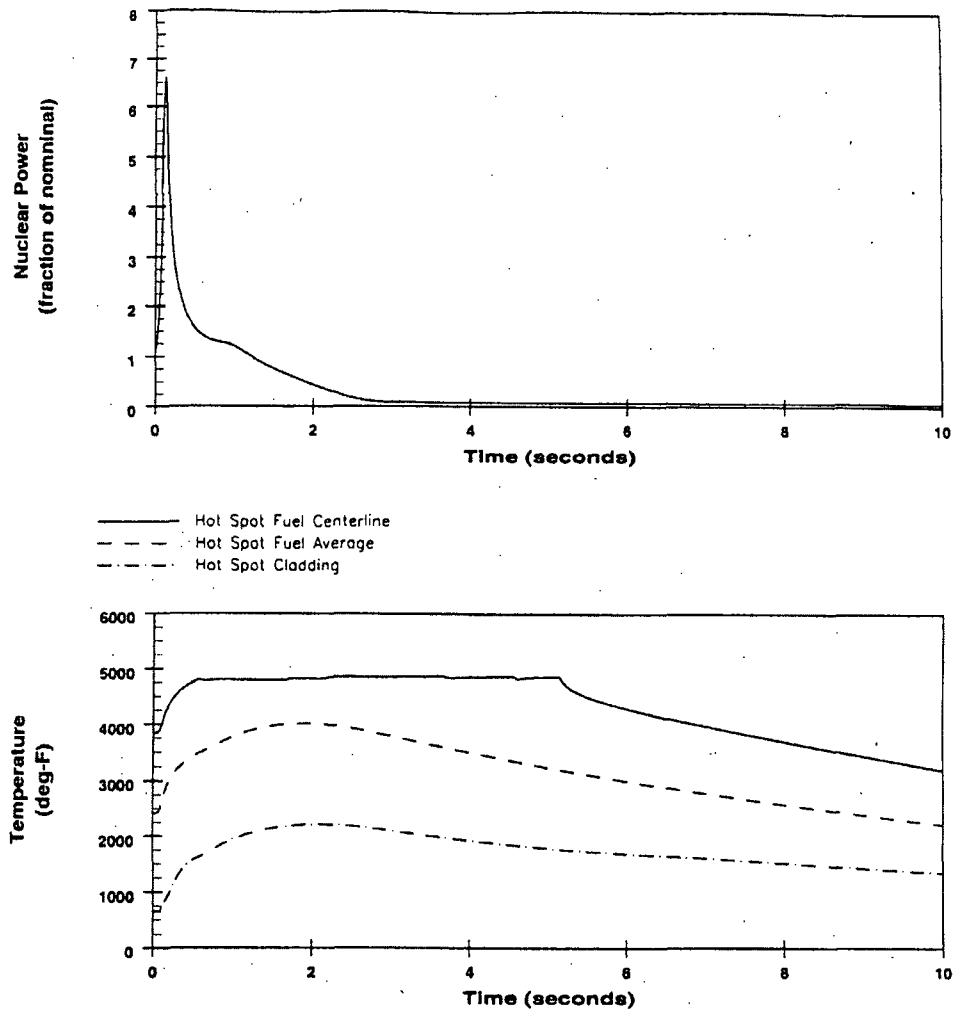


Point Beach Nuclear Plant
Units 1 and 2

RCCA Ejection Transient
End of Life
Zero Power

Figure 14.2.6-3

FIGURE 14.2.6-4
RCCA EJECTION TRANSIENT
END OF LIFE
FULL POWER



Point Beach Nuclear Plant
Units 1 and 2

RCCA Ejection Transient
End of Life
Full Power

Figure 14.2.6-4

14.3.5 Radiological Consequences Of Loss Of Coolant Accident

The results of analyses presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a loss-of-coolant accident result in calculated offsite radiological doses that do not exceed the limits specified in 10 CFR 100 and result in calculated control room doses that do not exceed the limits of 10 CFR 50 Appendix A GDC 19 (Reference 9). The calculated doses are summarized in Table 14.3.5-6.

Basic Events and Release Fractions

A large break LOCA has been analyzed to determine the thyroid and whole body doses (Reference 2). There are two release pathways considered in this analysis: (1) radioactivity which enters containment from the reactor core and is released due to containment leakage, and (2) radioactivity which is released to the environment via ECCS equipment leakage.

The event causing the postulated releases is a double-ended rupture of a reactor coolant pipe, with subsequent blowdown, as described in Section 14.3.4. As demonstrated by the analysis in Section 14.3.2, the emergency core cooling system, using emergency power, keeps cladding temperatures well below melting and limits zirconium - water reactions to an insignificant level, assuring that the core remains intact and in place. As a result of the increase in cladding temperature and the rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. For analysis purposes, the entire inventory of volatile fission products contained in the core is assumed to be released during the time the core is being flooded by the emergency core cooling system. Of this core inventory, 50% of the halogens and 100% of the noble gases are assumed to be released to the containment vessel atmosphere. Fifty (50) percent of the halogens are assumed to be released to the containment sump.

Containment Vessel Inventory and Release Rate

The calculations of total core inventories were consistent with the methods used in Regulatory Guide 1.4 (Reference 1). Operation of the core at 1549 MWt was assumed.

It is not expected that a significant amount of organic iodine would be liberated from the fuel as a result of a loss-of-coolant accident. This conclusion is based on the results of fuel meltdown experiments conducted by the Oak Ridge National Laboratory. The fraction of the total iodine which is released in organic forms is expected to be on the order of 0.2%, or less, since the rate of thermal radiolytic decomposition would exceed the rate of production.

Organic compounds of iodine can be formed by reaction of absorbed elemental iodine on contaminated surfaces of the containment vessel. Experiments have shown that the rate of formation is dependent on specific test conditions such as the concentrations of iodine and impurities, radiation levels, pressures, temperatures, and relative humidity. The rate of conversion of airborne iodine is proportional to the surface to volume ratio of the enclosure, whether the process is limited by diffusion to the surface or by the reaction rate of the absorbed iodine. The yields of organic iodine observed as a function of aging time in various test enclosures were extrapolated to determine the values for the Point Beach containment vessels, using the variation of the surface to volume ratios. The iodine conversion rates predicted in this manner did not exceed 0.005% of the atmospheric iodine per hour. For the purpose of calculating doses, it has been assumed that 4% of the iodine activity released to the containment during the accident is immediately converted to the organic forms (methyl iodine), and that the containment spray has zero effectiveness in cleaning up iodine in this form. The organic form of iodine (methyl iodine) is assumed to be removed only by radioactive decay and leakage. The actual effectiveness of the spray system in removing organic iodine is discussed in Appendix C.

The particulate and elemental iodine are removed from the containment atmosphere by the action of the containment sprays. The effectiveness of the spray system for elemental iodine removal is also discussed in Appendix C. For the Point Beach plant, a two loop design, an elemental iodine removal coefficient of 20 hr^{-1} was determined based on the model suggested in NUREG-0800 (SRP 6.5.2) . Credit is taken is taken for this removal mechanism until a decontamination factor of 200 in the elemental iodine inventory in containment has been reached. The particulate iodine spray coefficient of 6.02 hr^{-1} is also determined based on the NUREG-0800 model. Credit is taken for particulate iodine removal until containment spray has been terminated. At that time, a decontamination factor of approximately 11.2 in the particulate iodine inventory in containment has been reached.

The containment leak rate of 0.4% by weight of containment air per day (4.6×10^{-8} sec^{-1}) was assumed to be maintained throughout the first 24 hours, and a leak rate of 0.2% per day (2.3×10^{-8} sec^{-1}) was maintained for the remainder of the 30 day period.

ECCS Equipment Iodine Inventory and Leakage Rate

When Emergency Core Cooling System (ECCS) recirculation is established following the LOCA, leakage is assumed to occur from ECCS equipment located outside containment. It is also assumed that 50% of the total core iodine is in the sump water being recirculated. Hence, the ECCS equipment leakage results in the release of a significant amount of iodine activity to the outside environment. For this activity release path, no credit is taken for plateout of elemental iodine on containment surfaces or for iodine removal by the atmosphere filtration system in the primary auxiliary building (PAB). The PAB vent stack is the source of emission for the ECCS leakage assumed in these analyses. The iodine release from this path is conservatively assumed to be 100% elemental.

The control room dose analysis assumes that the ECCS equipment leaks at a rate of 400 cc/min. To demonstrate additional conservatism, the offsite dose analysis doubles the leak rate, and assumes 800 cc/min. These leak rates are conservatively assumed to continue at a constant rate from the time ECCS recirculation is established until 30 days following accident initiation. Ten percent of the iodine in the leakage is assumed to become airborne due to flashing.

There is no noble gas activity in the ECCS recirculation water.

Off-Site Inhalation Doses

The committed dose equivalent to the thyroid (CDE-Thyroid) resulting from activity leaking from the reactor containment vessel following the postulated loss-of-coolant accident was computed for the five iodine isotopes from the following expression.

$$D(x,T) = \int_0^T S(t)\, L(t)\, \chi/Q(x,t)\, B(t)\, dt$$

The source term for the five isotopes is:

$$S(t) = \sum_{j=1}^{5} C_j \; DCF_j \; [(1-\beta) \exp(-\lambda_j - \lambda_{si})\, t + \beta \exp(-\lambda_j)t]$$

The terms in the relationships are defined as follows:

t	=	time since reactor shutdown, hours
T	=	time period over which committed dose equivalent is accumulated, hours
x	=	distance, meters
D(x,T)	=	committed dose equivalent, rem
L(t)	=	containment leak rate, second^{-1}
χ/Q(x,t)	=	site Dispersion factor, second/m^3
B(t)	=	breathing rate, m^3/hour
C_j	=	activity of isotope j in the containment at reactor shutdown, curies
DCF_j	=	exposure-to-dose conversion factor for inhalation for isotope j, rem/curie
β	=	initial fraction of containment iodine inventory which is in the organic form, dimensionless
λ_j	=	radioactive decay constant for isotope j, hour^{-1}
λ_{si}	=	spray removal coefficient for inorganic iodine^{-1} including removal by condensation, hour^{-1}

The CDE-Thyroid due to the inhalation of the radioiodines was calculated for a range of values of the spray removal coefficient. The doses reported in Table 14.3.5-6 for the site boundary and the low population zone were based on a spray removal coefficients described in Table 14.3.5-4. The values of the containment leak rate and the fraction of organic iodine were discussed in the previous section. The breathing rates described in Table 14.3.5-2 are in accordance with NRC Regulatory Guide 1.4. The exposure-to-dose conversion factors for inhalation, described in Table 14.3.5-2, are based on ICRP Publication 30.

Site Dispersion Factors - Point Beach Nuclear Plant

The Site Boundary and Low Population Zone atmospheric dispersion factors (χ/Q) are described in Table 14.3.5-2.

The Control Room dispersion factors were calculated using the equations given in Reference 6, and are presented in Table 14.3.5-2. The equation for a diffuse source and point receptor is used. This equation is used when radioactivity is assumed to leak from many points on the surface of the containment building in conjunction with a single point receptor. The use of this equation is also appropriate for a point source / point receptor where the difference in elevation between the source and receptor is greater than 30 percent of containment height.

Using the methodology described in Regulatory Guide 1.145 and Murphy/Campe, the dispersion factor (χ/Q) value that is exceeded five percent of the time for PBNP was calculated using the hourly site-specific meteorological data measured during the 1991 through 1993 calendar years. For each hour of meteorological data, the corresponding dispersion factor (χ/Q) value at the control room ventilation intake distance from the containment building was calculated. These χ/Q values were then used to construct a cumulative probability distribution of χ/Q values for the site. This probability distribution is described in terms of probabilities of exceeding a χ/Q value during the total time.

Control Room Doses

The analysis for radiological consequences in the control room was based on operation of the control room ventilation system in Mode 4, which is unfiltered recirculation with filtered makeup. Other assumptions related to the calculation of control room inhalation dose are described in Tables 14.3.5-1 through 14.3.5-5.

The direct radiation dose due to the plume outside of the control room is calculated using the computer program QAD-CGGP (Reference 7). The inputs are consistent with those used in the large break LOCA dose calculations. It is assumed that a control room operator is located 10 feet inside the control room window for 75% of the occupancy time and 5 feet inside the control room window for 25% of the occupancy time (Reference 8).

The 30-day whole body dose from the radionuclides within the control room for the large break LOCA event is 1.37 rem. Therefore, the direct dose to operators from radiation outside the control room must be below 3.63 rem in order to remain within the 5 rem total whole body dose limit. With portable lead shielding placed in front of the control room door and window, the direct dose is approximated at 3 rem over the 30-day duration of the event. Thereby, the 5 rem whole body dose limit is met.

Off-Site Whole Body Doses

The analytical model used for cloud submersion doses assumes the leaking volatile fission products form a semi-infinite cloud, with concentration equal to the calculated centerline concentration. The expression used to calculate the submersion is as follows:

$$D\gamma(x,T) = \int_{o}^{T} S(t)\, L(t)\, \frac{\chi}{Q}\,(x,t)\, dt$$

The source term for the volatile fission products is:

$$S(t) = \sum_{j=i}^{7} C_j \; DCF_j \; e^{-\lambda_j t}$$

The terms in the equations are defined as follows:

t	=	time since reactor shutdown, hours;
T	=	time period over which the committed effective dose equivalent is accumulated, hrs;
x	=	distance, meters;
$D\gamma(x,T)$	=	committed effective dose equivalent, rem;
L(t)	=	containment leak rate, $second^{-1}$
$\chi/Q\ (x,t)$	=	site dispersion factor, seconds/cubic meter
C_j	=	activity of isotope j in containment at time 0, curies
DCF_j	=	the effective dose equivalent conversion factor for submersion for isotope j, rem-m^3/curie-hour;
λ_j	=	radioactive decay constant for isotope j, $hour^{-1}$

For calculation of the cloud submersion doses, the same containment inventories, leak rates, site dispersion factors, and basic release fractions were used as in the inhalation dose calculations.

Isotope	DCF (EDE - Submersion) (rem-m^3/curie-hour)
Kr-85	1.74E+00
Kr-85m	1.10E+02
Kr-87	5.25E+02
Kr-88	1.33E+03
Xe-133	2.25E+01
Xe-133m	1.99E+01
Xe-135	1.73E+02

The doses were calculated based on release fractions of a full core inventory per Reference 1, and the results are presented in Table 14.3.5-6. The contribution to the whole body doses from direct radiation from the containment vessel, reported in Section 11, is negligible.

References

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

2. Westinghouse Calculation Note CN-CRA-96-119, Rev. 0, "Point Beach Large LOCA Offsite and Control Room Dose Analysis for Replacement Steam Generator Program," dated February 1997.

3. Toner, D.F., and Scott, J.S., Fission Product Release from UO_2, Nuclear Safety, Vol. 3, No. 2, December, 1961.

4. Belle, J., Uranium Dioxide: Properties and Nuclear Applications, Naval Reactors, DRD of USAEC, 1961.

5. Eckerman, Keith F., Wolbarst, Anthony B., and Richardson, Allan C.B., Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, Federal Guidance Report Number 11, EPA-520/1-88-020, September 1988.

6. K.G. Murphy and K.M. Campe, "Nuclear Power Plant Ventilation System Design for Meeting General Criterion 19," 13th AEC Air Cleaning Conference.

7. QAD-CGGP, "A Combinatorial Geometry Version of QAD-P5A, A Point Kernel Code System for Neutron and Gamma-Ray Shielding Calculations Using the GP Buildup Factor."

8. WE Letter to NRC, NPL-97-0315, "Supplement to Technical Specifications Change Request 192," dated June 3, 1997.

9. NRC SER dated July 9, 1997, "Issuance of Amendments Re: Technical Specifications Changes for Revised System Requirements to Ensure Post-Accident Containment Cooling Capability."

TABLE 14.3.5-1
CORE ACTIVITIES[1]

Nuclide	Total Core Activity at Shutdown (Ci)
I-131	4.13 E7
I-132	5.92 E7
I-133	8.45 E7
I-134	9.30 E7
I-135	7.89 E7
Kr-85	5.07 E5
Kr-85m	1.13 E7
Kr-87	2.16 E7
Kr-88	3.0 E7
Xe-131m	4.41 E5
Xe-133	8.36 E7
Xe-133m	2.63 E6
Xe-135	2.30 E7
Xe-135m	1.60 E7
Xe-138	7.04 E7

[1] These core activities are based on a core power level of 1548.9 MWt. The activities were updated as a part of the Point Beach fuel upgrade program.

TABLE 14.3.5-2
DOSE CONVERSION FACTORS, BREATHING RATES, AND ATMOSPHERIC DISPERSION FACTORS

Isotope	Thyroid Dose Conversion Factors[2] (rem/curie)
I-131	1.07 E6
I-132	6.29 E3
I-133	1.81 E5
I-134	1.07 E3
I-135	3.14 E4

Time Period (hr)	Breathing Rate[3] (m^3/sec)
0 - 8	3.47 E-4
8 - 24	1.75 E-4
24 - 720	2.32 E-4

	Atmospheric Dispersion Factors[4] (sec/m^3)
Site Boundary	
0 - 2 hr	5.0 E-4
Low Population Zone	
0 - 8 hr	3.0 E-5
8 - 24 hr	1.6 E-5
24 - 96 hr	4.2 E-6
96 - 720 hr	8.6 E-7

Control Room[5]	Release from Containment	Release from Auxiliary Building
0 - 8 hr	3.0 E-3	1.7 E-3
8 - 24 hr	1.9 E-3	1.2 E-3
24 - 96 hr	1.2 E-3	6.7 E-4
96 - 720 hr	4.8 E-4	2.3 E-4

2 ICRP Publication 30
3 Regulatory Guide 1.4
4 Wisconsin Electric letter NPL 97-0041
5 Wisconsin Electric letter NPL 97-0276

TABLE 14.3.5-3
CONTROL ROOM PARAMETERS

Volume	65,243 ft^3
Unfiltered Inleakage	10.0 cfm
Total Flow Rate	19,800 cfm
Filtered Makeup	4,950 cfm
Filtered Recirculation	0 cfm
Filter Efficiency	
Elemental	95%
Organic (Methyl)	95%
Particulate	99%
Occupancy Factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
4 - 30 days	0.4

TABLE 14.3.5-4
ASSUMPTIONS USED FOR LARGE BREAK LOCA DOSE ANALYSIS
CONTAINMENT LEAKAGE

Power (102%)	1549 MWt
Iodine Chemical Species	
Elemental	91%
Methyl (Organic)	4%
Particulate	5%
Iodine Removal in Containment	
Instantaneous Iodine Plateout	50%
Containment Spray	
Start delay time	90 seconds
Injection spray flow rate	1190 gpm
Duration of injection spray	65 minutes
Spray removal coefficient	
Elemental	20 hr^{-1}
Particulate	6.02 hr^{-1}
Containment Net Free Volume	1.065 E6 ft^3
Sprayed Volume	475,000 ft^3
Unsprayed Volume	590,000 ft^3
Containment Mixing	
Containment Fan Coolers	
Start Delay Time	90 Seconds
Number of Units	2
Flow Rate per Unit	38,500 cfm
Containment Leak Rate	
0 - 24 hr	0.4%/day
> 24 hr	0.2%/day

TABLE 14.3.5-5
ASSUMPTIONS USED FOR LARGE BREAK LOCA DOSE ANALYSIS
ECCS EQUIPMENT LEAKAGE

Power (102%)	1549 MWt
Iodine Activity in Recirculation Water	50% Core Iodine Activity
Iodine Chemical Species	100% Elemental
Leakage Rate	
For Offsite Doses	800 cc/min
For Control Room Doses	400 cc/min
Leakage Duration	From start of recirculation through 30 day duration
Time of Recirculation Initiation	20 minutes
Sump Water Volume	197,000 gallons *
Iodine Flashing Fraction to Environment	10%

* Modifications have been made to both units to minimize hold up in the lower refueling cavity. This will provide a greater Sump Water Volume.

TABLE 14.3.5-6
LARGE BREAK OFFSITE AND CONTROL ROOM DOSES

1. Thyroid Doses

	***** Dose (Rem) *****		
	SB (0-2 Hr)	LPZ (0-30 Day)	CR (0-30 Day)
Containment Leakage	133.3	24.37	186.0
ECCS Equipment Leakage	57.12	37.0	106.7
Total	190.42[6]	61.37[1]	292.7[7]

2. Whole Body Doses

	***** Dose (Rem) *****		
Containment Leakage (IO)	SB (0-2 Hr)	LPZ (0-30 Day)	CR (0-30 Day)
Containment Leakage (NG)	0.72	0.07	0.012
ECCS Equipment Leakage	2.52	0.38	1.354
Total	0.24	0.06	0.004
	3.48	0.51	1.37

3. Control Room β-Skin Dose

	Dose (Rem) 30 Day
Containment Leakage (IO)	0.12
Containment Leakage (NG)	43.02
ECCS Equipment Leakage	0.04
Total	43.18

IO = Iodines
NG = Noble Gases

[6] NUREG-0800 Acceptance Criterion 300 rem thyroid
[7] NUREG-0800 Acceptance Criterion 30 rem thyroid. Assumes administration of 100 mg potassium iodide tablets will reduce calculated thyroid dose by a factor of 10.

FIGURE 14.3.5-1
SITE DISPERSION FACTORS
BUILDING WAKE FACTOR = $(1 \times 1640)m^2$

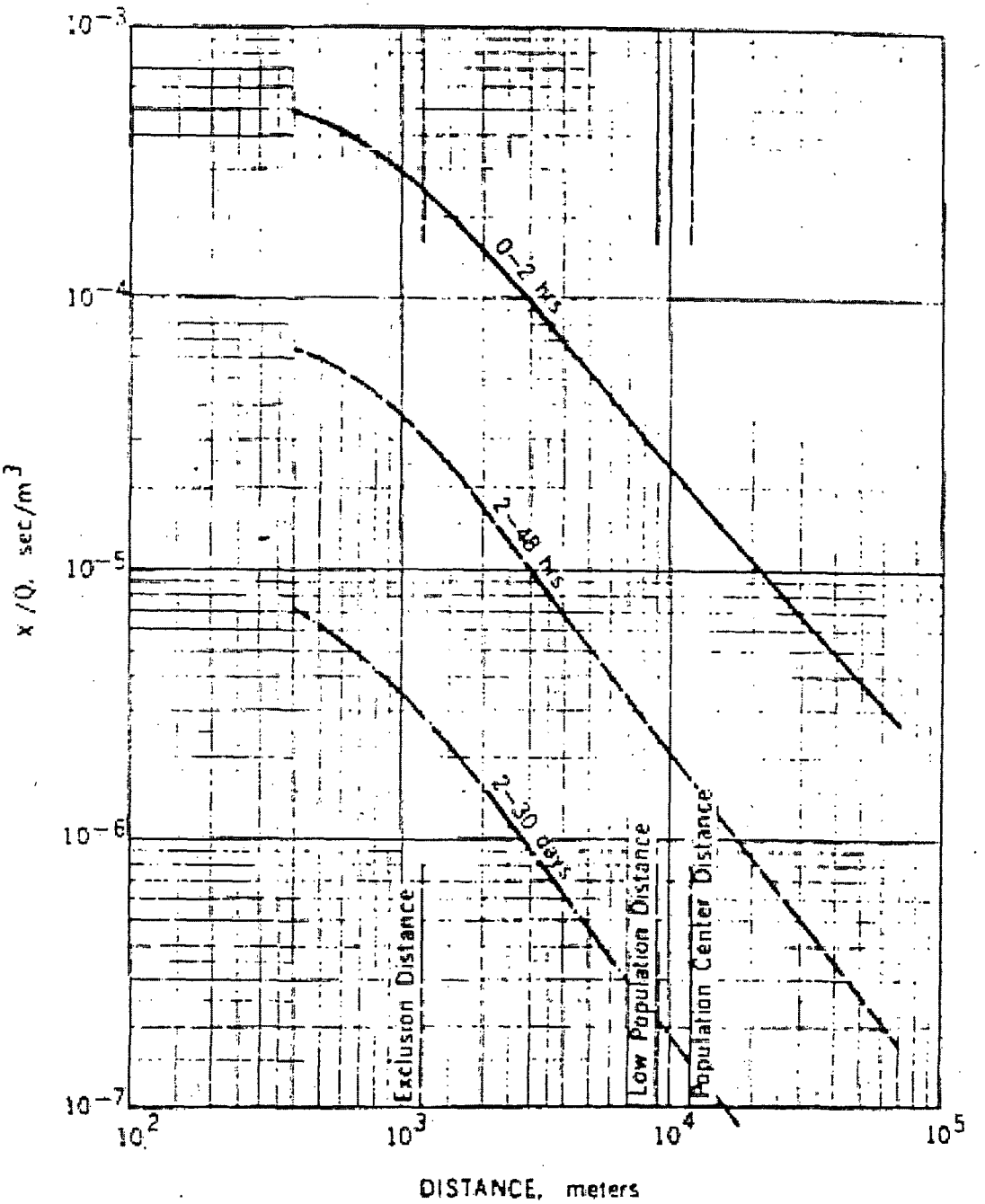
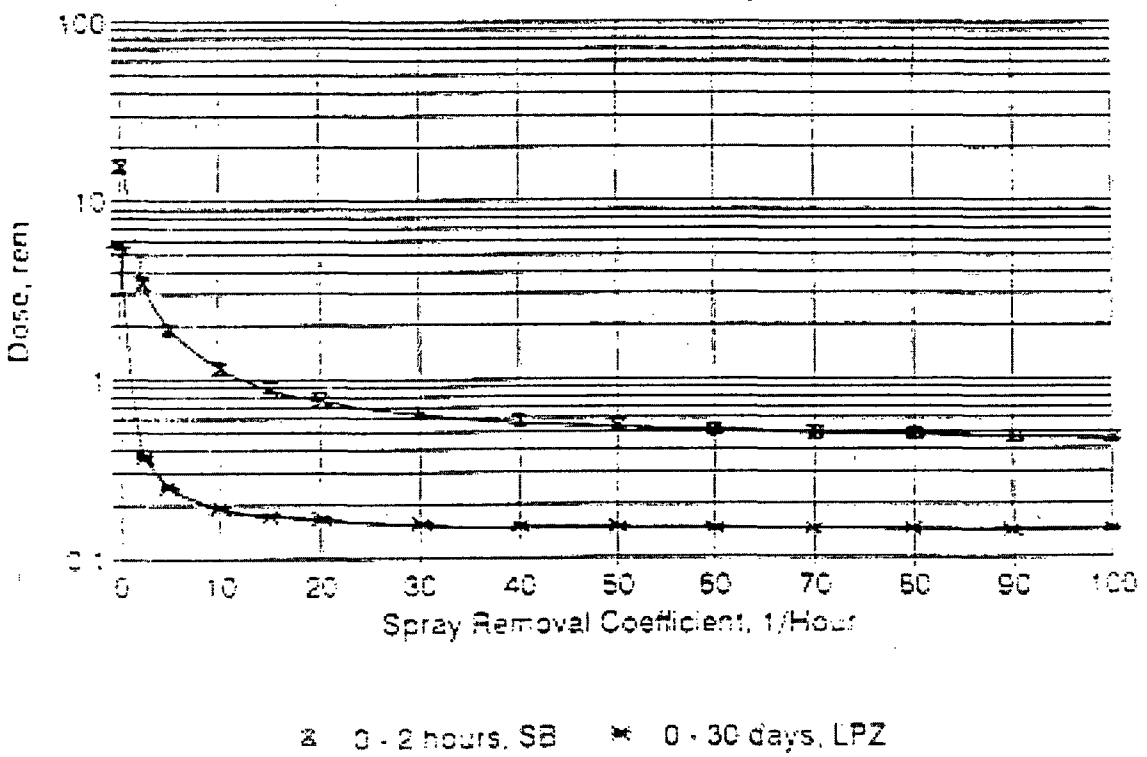


FIGURE 14.3.5-2
INHALATION DOSE AS A FUNCTION OF SPRAY REMOVAL COEFFICIENT (GAP ACTIVITY)



14.3.6 REACTOR VESSEL HEAD DROP EVENT

The analyses presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a postulated Reactor Vessel Head (RVH) drop event result in calculated offsite radiological doses that are well within the limits specified in 10 CFR 100, and result in calculated control room doses that do not exceed the limits of 10 CFR 50, Appendix A, General Design Criteria (GDC) 19. The calculated doses are summarized in Table 14.3.6-3.

To resolve questions pertaining to a postulated RVH drop event initiated as a result of the Unit 2 reactor head replacement in 2005, analyses were performed and submitted for NRC review and approval. Reference 1 is the Safety Evaluation (SE) documenting NRC acceptance of those analyses and is applicable to both units.

Reference 1 confirmed 15 regulatory commitments made by NMC associated with incorporating the Reactor Vessel Head Drop Event in the FSAR (Reference 2). The first 12 commitments pertain to requirements to be met or in place prior to initiating a reactor vessel head lift over a reactor vessel containing fuel assemblies. The commitments are as follows:

1. The reactor has been shut down for greater than 100 hours.

2. A Senior Reactor Operator will be stationed in containment during reactor vessel head-lift activities and will have communications capability with the control room.

3. The containment sump screen shall be installed and the flowpath for aligning RHR pump suction to the containment sump available.

4. A minimum borated water volume of 243,000 gallons shall be available for sump recirculation.

5. The containment equipment hatch will be on and bolted. Both personnel airlock door interlocks will be functional to ensure one door in each airlock is closed.

6. Containment purge supply and exhaust fans are off and associated containment isolation valves are closed when the reactor vessel head is suspended greater than 24 inches over the reactor vessel flange.

7. Other containment penetrations that allow containment atmosphere to communicate with the environment or the Primary Auxiliary Building atmosphere shall be closed.

8. The maximum allowable lift height for the reactor vessel head (i.e., 26.4 feet above the reactor vessel flange when over the fuel) shall not be exceeded.

9. Both safety injection trains shall be available.

10. Both residual heat removal trains shall be operable.

11. Technical Specification LCO 3.7.9, "Control Room Emergency Filtration System (CREFS)", and LCO 3.3.5, "CREFS Actuation Instrumentation," shall be met.

12. One standby emergency power source capable of supplying each 4.16 kV/480 V Class 1E safeguards bus on the associated unit shall be operable.

13. PBNP will incorporate an analysis of the RVH drop into the PBNP FSAR. This FSAR section incorporates that analysis.

14. PBNP will incorporate the PBNP method of NUREG-0612 Phase I compliance into the PBNP FSAR. FSAR Appendix A.3, "Control of Heavy Loads," addresses NUREG-0612 Phase I compliance

15. The Programmed and Remote (PaR) reactor vessel in-service inspection device will not be lifted over a core containing fuel assemblies. (Note: This restriction is identified both in this FSAR section and in FSAR Appendix A.3.)

14.3.6.1 Occurrences That Lead To The Initiating Event

While the potential causes of an RVH drop event are not specified in the NRC safety evaluations or the supporting submittals, such an event can be postulated to occur from mechanical failure of the crane hoist mechanism, cable failure, or RVH lift rig failure. The main hoist of each polar crane is equipped with two independent upper travel limit switches to prevent the possibility of a "two-blocking" incident. The two independent upper travel limit devices are of different design and are activated by independent mechanical means. These devices independently de-energize either the hoist drive motor or the main power supply. Since the upper travel limit switches on the containment polar cranes are independent, are tested, and operational restrictions limit upward travel, it was established in Reference 3 that the potential for an RVH drop event due to "two-blocking" (i.e., exceeding the physical upper travel limits of the crane) is negligible. See FSAR Appendix A.3 for additional discussion on "two-blocking."

14.3.6.2 Event Frequency Classification

The initiating event in this assessment is the drop of the RVH while it is suspended over the reactor vessel. The RVH is assumed to fall onto the reactor vessel flange, resulting in damage to the reactor vessel support structure.

NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," was written to address NRC Candidate Generic Issue 186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants." Crane operating history from 1968 through 2002 was reviewed as part of this report to provide a risk assessment associated with lifts of Very Heavy Loads (VHL). The risk analysis included in NUREG-1774 considers VHL lifts for any crane at any operating nuclear station. The analysis considers a postulated drop of load at any point during the movement of a load from the initial lift until set-down.

The probabilistic analysis contained within NUREG-1774 is primarily concerned with the probability of a VHL drop at an operating commercial nuclear power plant. A VHL is defined as any load over 30 tons. The generic probability for any VHL drop is given as 5.6E-5 per lift. This value is based upon three (3) drops per 54,000 VHL lifts.

Reference 5 established that a postulated RVH drop meets the frequency classification of an infrequent incident (i.e., an incident that may occur during the lifetime of the plant).

14.3.6.3 Sequence of Events

The analyzed event is a concentric drop of the RVH onto the reactor vessel flange from a height of 26.4 feet. This was determined to impart the maximum credible impact loads on the reactor vessel and supporting structures. The resultant impact displaces the reactor vessel downward. Downward movement of the vessel creates the potential for damage to piping and tubing directly or indirectly connected to the reactor vessel, thereby creating a potential for a decrease in reactor coolant inventory.

Upon impact with the vessel flange, the kinetic energy of the vessel head is partially dissipated and partially transferred to both the head (rebound) and the vessel through an elastic/plastic collision. The impact forces, if high enough, can lead to yielding of the vessel supporting structures and/or attached piping.

After the head and vessel have come to rest, decay heat removal can be maintained by one or both RHR trains and/or injection by the SI pumps. These same systems are adequate to makeup for any reactor coolant system (RCS) leakage that may occur from damaged Bottom Mounted Instrumentation (BMI) guide tubes. Damage to the point of rupture or shearing of other connected piping, including the main RCS loops, pressurizer surge line, core deluge lines, accumulator dump lines, normal charging and cold leg SI Lines), etc. are not expected.

Should RCS leakage be substantial, RHR suction can be re-aligned to the accident sump to ensure a continued cooling and makeup once available suction sources are depleted.

The mechanical shock of the impact is also postulated to result in damage to all fuel assemblies which leads to a complete fission product gap release. The resulting release of radioactivity must be mitigated by containment to maintain control room and off-site exposures within acceptable limits. Establishing a stringent containment closure prior to lifting the RVH ensures that this mitigating function is maintained.

14.3.6.4 Plant Characteristics Considered in the Safety Evaluation

To demonstrate the capability of the reactor vessel, RCS, and supporting systems and structures to sustain a postulated RVH drop event, two complementary inelastic structure and piping system analyses were performed (References 7 and 8). A RVH drop is postulated to occur during refueling when the head is manipulated above the reactor vessel. The RVH is assumed to fall concentrically onto the reactor vessel. Established administrative controls limit the maximum RVH drop height to 26.4 feet. This drop height has been utilized in the analyses discussed below.

The Sargent & Lundy (S&L) analysis (Reference 7) evaluated the reactor vessel and vessel support behaviors using a finite element model. The Westinghouse analysis (Reference 8) evaluated the plastic deformation that may occur to connected RCS piping based on specified bounding reactor vessel displacements.

S&L Finite Element Analysis

This analysis considers a flat vertical impact of the new RVH, which weighs 200,000 lbs, dropping from a height of 26.4 feet onto the reactor vessel flange. This analysis also includes an evaluation of the structural integrity of supporting elements in the load path, and predicts the vertical downward displacement of the reactor vessel.

The load path consists of the reactor vessel, reactor vessel supports at the four RCS nozzles and two brackets under the RHR core deluge nozzles, the support girder box frame, and the six pipe columns and their supports, which rest on the concrete foundation. The reactor coolant system (RCS) piping provides additional stiffness to the reactor vessel nozzles under vertical impact loading, and also transfers a portion of the impact load to the steam generator (SG) and the reactor coolant pump (RCP) support structures under a postulated RVH scenario. The concrete and embedded reinforcing bar located between the support girder and the concrete foundation under the support columns is not considered to provide any vertical support, even if the predicted deflection of the vessel could result in contacting the concrete.

The analysis models used are static analysis models for stiffness calculations of various components and substructures, and a dynamic impact model. The finite element analyses are performed using the ANSYS computer code.

The static analysis models include:

(1) A detailed model of reactor vessel flange and reactor vessel shell below the flange, including a nozzle resting on a supporting shoe.

(2) A similar detailed model of reactor vessel flange and reactor vessel shell below the flange with a support bracket resting on a supporting shoe.

(3) A detailed model of the hexagonal girder box frame supported by six pipe columns at the vertices.

(4) Piping models for the RCS hot legs and cold legs.

These models are used to construct static load-displacement diagrams for all steel components that are within the impact load path. Static vertical displacement is applied to the components uniformly and a reaction force is calculated to construct the force-displacement diagram of the affected components. In the static analysis, non-linear material properties are modeled with a strength increase factor of 10 percent to account for the strain rate effects due to the dynamic impact. The large deformation analysis option was selected to account for potential buckling and yielding in the structural components along the impact load path.

The results of the static analysis are used as part of the input for dynamic analysis. In calculating the stiffness of RCS hot leg or cold leg, two bounding cases are analyzed:

1) A fixed boundary condition is used at either the SG location or the RCP location.
2) A pinned boundary condition is used at the SG location or the RCP location.

In both cases, the pipe axial movement is released to account for the potential horizontal movement of the SG or the RCP.

The dynamic impact model consists of a two-mass model with springs and dash-pot in a vertical configuration. The top mass represents the falling head, and the bottom mass represents the target reactor vessel model supported by various springs, which represent the stiffness of the nozzle/bracket support, the girder box frame/column supports, and the RCS piping.

In the dynamic impact analysis, an impact damping of 5% of the critical damping is used. This assumption is judged to be reasonable for this application in consideration of:

1) energy loss due to plastic damage at the impact surface between the RVH and the reactor vessel flange;
2) energy loss due to imparted damage to six lateral supports for the hexagonal girder box frame; and,
3) energy loss due to local damage to the liner and concrete crushing at the top of the six support columns.

Results of the dynamic transient analysis indicate that the maximum dynamic downward displacement of the reactor vessel is 2.72 and 3.20 inches for cases 1 and 2 respectively. These displacements are both less than the 3.375" necessary before the hexagonal girder box frame would come into contact with the concrete "shelf", and this is consistent with the assumption that the concrete shelf does not provide any resistance to downward motion.

Using the limiting downward displacement of 3.2", the maximum Von Mises stress in the nozzle due to membrane plus bending is less than the ASME Boiler and Pressure Vessel Code, Section III, Appendix F allowable stresses for membrane stress intensity of 0.7 S_u. Similarly, the Von Mises stress in the reactor vessel support brackets is also less than 0.7 S_u.

The S&L analysis also evaluated the maximum impact load on the column foundation, and the capability of the concrete shelf to provide lateral support for the stability of the support columns (i.e. to limit buckling) located within the shelf and found the results acceptable.

Westinghouse Plastic Analysis of RCS Loop Piping

The evaluation consisted of a plastic analysis of the PBNP reactor coolant loop piping for a downward vertical displacement of the reactor vessel nozzles. Two displacements were analyzed: (1) a 4-inch displacement, which bounds the displacement calculated by the S&L model, and (2) a 6.5-inch displacement, which represents the maximum possible displacement of the reactor vessel nozzles before the RCS piping comes in contact with the biological shield wall.

The results of the analysis were compared to the criteria specified in the 1998 Edition of ASME Code, Section III, Appendix F, Paragraph F-1340. The criteria allow for large RCS loop piping deformations, with the intent that violations of the RCS pressure boundary do not occur.

The analysis uses an ANSYS finite element model of the hot and cold legs. The hot and cold legs are fixed at both ends (the reactor vessel nozzles and the SG or RCP nozzles). Each leg was modeled as a straight run of piping with one elbow. The hot and cold leg material properties were represented by a piece-wise linear stress-strain curve. Two sets of material properties were used to represent the upper and lower bound properties of the piping and elbow materials.

The results of the analysis indicate that the maximum calculated stress intensity in the hot and cold leg piping is within the ASME Code, Section III, Appendix F limit of 0.7 S_u for general primary membrane stress for the 4-inch reactor vessel nozzle displacement. Since the 4 inch reactor vessel nozzle displacement bounds the maximum calculated vessel displacement predicted from the S&L model, there is reasonable assurance that the pressure boundary integrity of the RCS loop piping will be maintained in the event of a postulated RVH drop.

The results also indicate that the 0.7 S_u limit is exceeded for the cold leg for a 6.5-inch vessel nozzle displacement. The maximum stress intensity was calculated in the cold leg elbow. While the calculated stress intensity exceeds the ASME Code general primary membrane stress intensity limit, it is concluded that loss of the RCS piping pressure boundary integrity would not be expected even if the vessel nozzle displaced 6.5 inches. This is because the maximum calculated stress intensity is still well below the material ultimate strength.

Analysis of Reactor Vessel Deflection

Based on the Sargent & Lundy FEA provided in Reference 7 and the Westinghouse analysis provided in Reference 8, the following bounding conditions apply:

Following the postulated RVH drop, using a conservatively estimated RVH weight of 200,000 lbs (Unit 1), the reactor vessel deflection would not exceed 3.36 inches. This calculated deflection is slightly greater than the Unit 2 calculated vessel deflection due to the conservative weight assumed and slight dimensional differences between units. RCS piping remains intact following the postulated reactor vessel deflection.

The impact of the postulated reactor vessel deflection on the attached RCS piping was assessed. This assessment was performed by Westinghouse and is documented in Reference 7. Westinghouse performed an analysis for a 4-inch deflection, which bounds the projected reactor vessel deflection. The results of the analysis show that stress values are less than the more restrictive criteria of $0.7S_u$ specified in ASME Section III Appendix F. In addition, a second case to analyze a deflection value of 6.5 inches, which is equivalent to the gap that exists between the RCS piping and the shield wall, was conducted. The results of this analysis yielded stress values of greater than $0.7S_u$ but did not predict failure of the RCS piping.

The combined results of the Sargent & Lundy and the Westinghouse analyses show that the damage from a RVH drop would not result in a loss of decay heat removal. Based on these results, it was concluded that adequate reactor core cooling and makeup capability would be maintained following the expected deflection of the reactor vessel from a postulated RVH drop.

Piping attached to the reactor coolant system (RCS) was not modeled or specifically analyzed for deflection and stress values as a result of the vessel deflection from a RVH drop. Based on the ability to analyze and demonstrate RCS piping acceptability for a bounding deflection of 4 inches, it was determined that the attached piping would also be acceptable. This conclusion was based upon the fact that all connections to the RCS piping are outside of the biological shield wall; thus, the deflection would be much less than the total deflection of the RCS piping. In addition, the attached piping is of smaller diameter and is more flexible. The main connections to the RCS, credited for maintaining core cooling and makeup following a RVH drop, are the residual heat removal (RHR) lines, cold leg safety injection (SI) injection lines and charging. The RHR suction and return lines are 10-inch lines; the cold leg SI flow path is through the 10-inch SI accumulator injection line connected to the RCS.

Charging and auxiliary charging are connected through a 3-inch and 2-inch line to the RCS. The 10-inch connections are the closest connections of concern to the reactor vessel, with one exception, and would therefore experience the greatest relative deflection. The only exception is that the Unit 1 Auxiliary Charging line is 10 inches closer to the reactor vessel than the corresponding Safety Injection line on the "B" cold leg. Since the Auxiliary Charging line is a 2-inch line with greater flexibility than the 10-inch SI line, the focus was on addressing the SI lines. For Unit 1, the ratio of the distance from the reactor vessel to the steam generators or reactor coolant pumps would yield a deflection of approximately 20 percent, or less, of the total vessel deflection. For a vessel deflection of 3.36 inches, the deflection at the connection would be approximately 0.67 inches.

In Unit 1, the shortest horizontal piping run from the 10-inch connections at the cold legs to the first vertical support (which is a spring hanger), is greater than 6 feet. The shortest vertical run is approximately 10 feet (on the opposite cold leg). Both connections have horizontal offsets that decrease their stiffness in the vertical direction. The shortest horizontal run to an anchor is greater than 14 feet with an intervening vertical loop.

The RHR return line connects to the SI accumulator injection line over 22 linear feet from the B loop cold leg connection. The condition is very similar for the RHR suction line connection to the A hot leg. The distance to the closest anchor is greater than 13 feet with an intervening vertical loop containing an additional 30 feet of piping.

In each case, the total linear distance between anchors for the attached piping is greater than the worst RCS piping case, and that case was shown to be acceptable for a deflection of 4 inches. Based on this, the added flexibility of smaller diameter piping and an equivalent deflection of approximately 0.67 inches, it was determined that a detailed analysis of the connected piping was not necessary.

Additionally, the integrity of the two 6-inch core deluge lines was evaluated based on comparing the section properties and applicable pipe spans to the RCS piping. This comparison, coupled with the fact that the core deluge lines are more flexible than the RCS piping, leads to the conclusion that the integrity of the core deluge lines are bounded by the assessment for the RCS piping.

Bottom-Mounted Instrument (BMI) Tubes

As a result of the predicted maximum dynamic downward displacement of 3.2 inches for the reactor vessel, and recognizing the potential impact between the BMI tubes and the floor (clearance varying between 1" and 4.5"), it was conservatively assumed that all 36 tubes are severed. Therefore, the structural integrity of the BMI tubes is not considered in the structural integrity analysis.

14.3.6.5 Protective System Actions

Core Cooling Configuration

The emergency core cooling system (ECCS) and normal core injection paths remain available during the postulated event. Core cooling water remains available to ensure adequate cooling and makeup is maintained to remove decay heat and keep the core covered.

Upon exhaustion of the Refueling Water Storage Tank (RWST) inventory, the residual heat removal (RHR) pumps would be realigned to take suction from the containment sump; with the safety injection pump(s) drawing from the RHR pump discharges as needed. This provides assurance that core cooling and makeup can be maintained for a prolonged period.

For Low Temperature Overpressure Protection of the reactor vessel, Limiting Condition for Operation (LCO) 3.4.12 requires one train of safety injection to be disabled when the RVH is installed on the vessel. Prior to head installation, one train of safety injection is configured to prevent inadvertent start and pressurization once the RVH is installed in order to satisfy this requirement. PBNP will maintain the second train as available using administrative controls currently defined in the shutdown safety assessment procedure. This procedure defines the safety assessment and risk management process used to comply with 10 CFR 50.65(a)(4) of the Maintenance Rule. In order to maintain the second train as available during head lift operations, PBNP has specified administrative requirements which will ensure prompt recovery if required. Pre-briefed and stationed operators would take the necessary local manual actions to ensure a timely recovery of the second train as necessary.

In the event of a RVH drop, operators will first be alerted by reports from personnel who witnessed the event. A Senior Reactor Operator (SRO) will be stationed inside containment during all such lifts. The function of the SRO is to communicate the occurrence of a RVH drop to the control room.

The operators will assess damage using indications available to them in the control room. The mitigating strategy will retain core cooling and makeup using RHR, charging and safety injection flow paths. Minimum equipment availability is established as prescribed in the regulatory commitments listed at the beginning of FSAR 14.3.6. Based on the need for an assured makeup capability, both trains of RHR and SI are required to be operable and available, respectively, during RVH lifts above a vessel containing irradiated fuel.

As an upper bounding scenario, all bottom mounted instrument (BMI) tubes are postulated to sever. The resultant inventory loss from the RCS is postulated to be approximately 300 gpm. This value is based on all 36 BMI tubes being severed in a manner that causes minimal flow restriction from the break and the resultant gravity fed flow of RCS water through the tubes' nominal interior diameters. The boiloff rate is minor when compared to the loss because complete failure of all 36 BMI tubes and is well within the capacity of a single SI or RHR pump. (Each RHR pump is rated for 1560 gpm at a design head of 280 ft, and an SI pump is rated for 700 gpm at a head of 2600 ft).

In the event of a loss of coolant event during cold shutdown, procedure controls are established to ensure adequate core inventory and cooling are maintained. The procedure performs the following:

1) Check RHR Pump conditions and secure, if required, due to system voiding
2) Establish charging from the RWST to maintain RCS inventory
3) Establish safety injection flow from the RWST via one SI pump, if necessary
4) Verify adequate injection flow based on inventory and RCS temperature
5) Establish low head injection flow, if required, based on temperature
6) Establish containment sump recirculation, if required, based on RWST level.

Time to boil curves provided in SEP-1, "Degraded RHR System Capability," show that the time to boil, 100 hours after shutdown as assumed in the current dose assessment, with RCS level at reduced inventory and starting at 140°F, is approximately 18 minutes. With the RCS level at one foot below the flange, which is the procedural requirement for lifting and setting the RVH and the same conditions as above, the time to boil is just over 23 minutes. Based on simulator validation of steps in SEP 2.3, "Cold Shutdown LOCA," the time to inject water into the RCS using SI pumps is approximately 10 minutes. Thus, adequate time is available to diagnose the problem, enter the associated shutdown emergency procedure, and establish makeup flow following a postulated RVH drop that results in RCS leakage.

Upon exhaustion of the minimum committed 243,000 gallon suction source (nominally the Refueling Water Storage Tank, RWST)[1], the RHR pump(s) can be realigned to take a suction from the containment sump. Accordingly, this accident sump suction path must be maintained available during movement of the RVH.

In addition to the provision of in-depth makeup capability, containment is required to be closed with the purge supply and exhaust fans off and the associated penetrations closed (either by valves or their equivalent), and personnel airlock door interlocks are functional (to ensure that at least one door at each airlock is closed) prior to movement of the RVH.

[1] This suction source is 243,000 gallons available for recirculation. This would be in addition to an inventory necessary to fill the lower refueling cavity. This is because the volume in the lower refueling cavity is isolated from the containment sump during head lift activities, and would not be available for dilution and hold-up of the source term inventory. Therefore, if the lower cavity is not flooded at the time of the lift, the available suction source must be 243,000 gallons *plus* the volume needed to fill the lower cavity.

14.3.6.6 Core and System Performance

The postulated RVH drop occurs with the reactor in cold shutdown with shutdown margin assured by a combination of inserted control rods and boron concentration. The drop is not postulated to result in changing the core geometry, and as such, no explicit evaluation or analysis of nucleonics has been performed.

System performance is addressed in section 14.3.6.5 (Protective System Actions) above.

14.3.6.7 Barrier Performance

The primary barrier against the adverse consequences of an RVH drop is prevention. The measures in place to minimize occurrence of such an event are described in detail earlier in this section.

The fuel pins are suspended by friction in the grid strap assemblies, and are relatively stiff in the axial direction. In addition, there is some axial spacing between the upper fuel nozzle block and the ends of the fuel pins. As a result, the pins can slide axially to accommodate some axial shock loading of the fuel assembly. Therefore, fuel clad failure is not anticipated to occur due to an RVH drop event. Nonetheless, to provide a bounding accident scenario, a failure of 100% of the fuel cladding is assumed by the radiological dose assessment. This assumed failure releases all of the clad gap inventory, but since the fuel remains covered, further degradation (e.g., Zr-water reaction, fuel melt) does not occur, and the ceramic fuel is assumed to retain all additional radionuclide inventory.

As previously discussed, a gross failure of the RCS pressure boundary is not expected. Some leakage is assumed to occur due to the assumed severance of BMI tubes. This leakage is well within the makeup capability of the SI and/or RHR systems, and this prevents further fuel damage due to uncovery.

Lastly, containment closure is a pre-condition for movement of the RVH over irradiated fuel (an exception is when the head is less than 24" above the vessel flange, when opening of the purge supply and exhaust valves and initiation of purge is permitted). As such, the containment comprises the final barrier against release of radioactive materials.

14.3.6.8 Radiological Consequence

An evaluation was performed to assess the offsite and control room dose consequences following a RVH drop for either unit. The event sequence assumes that the RVH drops onto the vessel causing fuel cladding damage to all of the fuel assemblies in the core, which results in a gap release. In addition, damage to the BMI tubes is assumed such that coolant is lost through these penetrations. Initial makeup of the RCS to the vessel is via suction from the RWST to the safety injection pumps, RHR pumps, or charging pumps. Once the RWST volume is exhausted, the RHR system is realigned to recirculate the coolant in the containment sump to maintain the core sub-cooled.

The dose evaluation for the RVH drop accident scenario described above uses a combination of the input assumption guidance for a design basis loss-of-coolant accident (LOCA) and a fuel handling accident (FHA), as they apply to the accident scenario. There is currently no explicit guidance for assessing the dose consequence of the RVH drop (or any heavy load drop) that results in an uncontrolled loss of coolant in addition to fuel damage.

For purposes of providing a bounding source term, the RVH drop is assumed to result in clad damage to 100% of all fuel assemblies, such that a complete gap release occurs.

Prior to moving the RVH, the following conditions are assumed to be in effect:

- Reactor has been shut down for a minimum of 100 hours;
- Containment equipment hatch and personnel airlocks are closed (equipment hatch on and bolted, one access door closed in each airlock, interlocks functional);
- Purge supply/exhaust system fans are off and isolation valves closed;
- Other containment penetrations that allow containment atmosphere to communicate with the environment or the PAB atmosphere are closed.

There are two possible release paths for a postulated RVH drop at PBNP:

- Containment - Through open penetrations or leakage through containment barriers,
- ECCS leakage - Due to eventual operation of the recirculation mode of the RHR system.

Since the release paths considered for the RVH drop are the same as those for the FSAR 14.3.5 LOCA radiological analysis, the dose consequences post-RVH drop are determined by scaling the FSAR LOCA dose consequences for the RVH drop source term and sump volume (dilution). Additionally, the control room operator doses are scaled to account for a higher unfiltered in-leakage into the control room and an assumed increased emergency core cooling system (ECCS) leakage rate. The scaling factor for each input assumption is determined by a ratio of the RVH drop input assumption and LOCA dose input assumption. The input assumption scaling factors are applied to the LOCA dose values to estimate the RVH drop dose consequence.

These release paths were evaluated using the guidance provided in Regulatory Guide (RG) 1.195. The guidance is appropriate for PBNP since 10 CFR 100 is the licensing basis for all of the design basis accident analyses except the fuel handling accident, which is licensed to 10 CFR 50.67. (NRC safety evaluation dated April 2, 2004, Reference 3).

Acceptance Criteria

The guidance in RG 1.195 does not provide specific accident offsite dose consequence criteria for the heavy load drop event. However, NUREG-0612 does categorize the heavy load drop event to be in the same class of limiting faults for which the radiological dose acceptance criteria are stated to be "well within" 10 CFR 100, meaning 25% of the 10 CFR 100 limits. The FHA is within this class of accidents, which generally would also be used to assess a heavy load drop such as RVH. Therefore, the offsite dose criteria are 6.3 rem whole body and 75 rem thyroid as listed in Table 4 of RG 1.195. The PBNP control room dose criteria are 5 rem whole body, and 30 rem thyroid as provided by 10 CFR 50, Appendix A, General Design Criteria (GDC) 19 and clarified by NUREG-0800 Section 6.4.

Containment Release Evaluation

The RVH drop dose assessment assumes that containment closure is established prior to occurrence of the event. Containment closure is defined as the containment equipment hatch and personnel airlocks are closed (equipment hatch on and bolted, one access door closed in each airlock, interlocks functional), the purge supply/exhaust system fans off and the isolation valves closed, and all penetrations that allow containment atmosphere to communicate with outside or PAB atmosphere closed.

The postulated RVH drop does not result in pressurization of the containment building. This is because the event occurs while the reactor coolant system is open to the containment building atmosphere and sufficient RCS makeup is available to provide cooling to the core. Heat is removed from containment by makeup to the RCS and containment sump recirculation. Since the containment configuration meets Footnote 2 to Position 5.1 of Appendix B to RG 1.195, and there is no pressure differential induced by the accident, there is no release via containment leakage.

ECCS Leakage Release Path Evaluation

The radiological dose assessment for the RVH drop assumes that the impact of the RVH onto the vessel results in cladding damage to 100% of the fuel assemblies. The event scenario also assumes complete severance of the BMIs such that there is an uncontrollable loss of coolant. Following confirmation of the event by Operations, makeup to the RCS is initiated to ensure that the core is covered and sub-cooled. Just prior to the exhaustion of the RWST inventory, the RHR pumps would be realigned to take suction from the containment sump; with the SI pump(s) drawing from the RHR pump discharges as needed. This provides assurance that core cooling and makeup can be maintained for a prolonged period. It is assumed that during the recirculation phase of the accident, this system is leaking containment sump coolant at a rate twice the limits of the Leakage Reduction and Preventive Maintenance Program for the emergency core cooling system (ECCS). No release during the injection phase is assumed.

Although the RVH drop results in a LOCA, the damage to the fuel assemblies is not driven by a thermal-hydraulic event but is assumed to occur due to impact of the head. This event occurs after the reactor has been shutdown and while the reactor vessel is open to the atmosphere. The accident scenario assumes that operations responds to an RVH drop by initially injecting borated coolant and eventually placing the RHR system in containment sump recirculation mode to provide long term decay heat removal. Since the accident occurs at temperatures and pressures well below operating levels and the accident mitigation strategy ensures that the core is covered and cooled, no additional fuel damage would occur. Therefore, the non-LOCA gap fractions provided in Table 2 of RG 1.195 are applicable to the RVH drop event since no additional release from the fuel (e.g., due to fuel melt) will occur. RG 1.195 provides a larger gap fraction for I-131 than for the other isotopes of iodine. Therefore, this evaluation conservatively applied the I-131 gap fraction of 8% to all isotopes of iodine.

The amount of activity released from the gap is determined from the total core inventory assumed for the LOCA analysis adjusted for the decay time of 100 hours from shutdown and the nuclide gap release fractions. The LOCA core inventory at shutdown can be found in PBNP FSAR Table 14.3.5-1. For the ECCS leakage path, all of the gap activity of iodine is assumed to be retained in the coolant while the noble gases are not retained in any appreciable amount in the coolant. Therefore, consistent with RG 1.195, Appendix B, Position 3, the evaluation did not consider a noble gas release through ECCS leakage. The source term for the RVH drop in the coolant (i.e., the sump source term) is provided in Table 14.3.6-1 below. The LOCA sump source term, which is one-half the total core inventory, is also provided. The source terms are based on power operation at 1549 MWt (1540 MWt licensed power plus 9 MWt calorimetric uncertainty).

The amount of coolant available for recirculation is equal to the amount of coolant that is injected. It is assumed that 243,000 gallons of borated coolant is injected into the vessel. The volume of coolant initially in the vessel and RCS is not credited for determining the dose consequence. The sump volume credited in the RVH drop analysis is larger than the value assumed in the LOCA analysis. A detailed discussion regarding the impact of increase in volume on the dose consequences is provided below.

The offsite and control room doses due to the RVH drop ECCS leakage are estimated by adjusting the LOCA ECCS leakage doses for the resulting RVH drop source term (Ci) and sump volume (gallons). Additionally, the control room dose estimates for the RVH drop take into account the ECCS leakage rate factor of two multiplier (Position 4.2 of Appendix A of RG 1.195) and a higher unfiltered in-leakage rate. The offsite LOCA ECCS leakage doses already take into account the factor of two multiplier for the ECCS leakage rate and are not impacted by unfiltered in-leakage. No other adjustments are made on LOCA ECCS leakage doses, other than those discussed above. Therefore, all other input assumptions used in the LOCA radiological consequence analysis are, in effect, used in the RVH drop ECCS leakage analysis.

Since dose is directly proportional to the amount of activity present, scaling factors can be used to estimate the dose due to RVH drop ECCS leakage. The scaling factors for the source term is listed in Table 14.3.6-2 and calculated by dividing the LOCA ECCS sump source term by the RVH drop sump source term found in Table 14.3.6-1.

The scaling factor for the source term accounts for the change in total activity present in the sump coolant as compared to the LOCA analysis. The I-131 sump source term scaling factor is used to adjust the dose due to the fact that it minimizes the dose reduction.

The activity in the sump is released to the environment via leakage from the ECCS during the containment sump recirculation phase of the accident. The activity released over time, post-sump recirculation (Ci), is directly proportional to the sump concentration (Ci/cc) multiplied by the leakage rate (cc/min). Increasing the sump volume reduces the sump concentration, which directly reduces the activity released. The scaling factor representing the increase in sump volume available for recirculation is calculated by dividing the LOCA sump volume of 197,000 gallons by the RVH drop sump volume of 243,000 gallons. The scaling factor for the sump volume accounts for the change in total activity released to the environment as compared to the LOCA analysis.

The control room operator doses are further scaled to account for an increase in unfiltered in-leakage and an assumed increase in the ECCS leakage rate. In order to assess the impact of the measured unfiltered in-leakage value on the control room operator dose post-RVH drop, a review of the constituents of the control room dose via the ECCS leakage pathway for PBNP is provided. The control room operator inhalation and whole body dose (due to activity internal to the control room) for PBNP is driven primarily by the amount of activity that passes through the control room ventilation filter into the control room. As described in FSAR Section 9.8, the control room emergency filtration system (CREFS) is actuated by a high radiation signal from control room area monitor RE-101, a high radiation signal from noble gas monitor RE-235 (located in the supply duct to the control room), or manually

from panel C67. When the CREFS is in operation (referred to as mode 4), it is assumed that 4950 cfm of outside air is supplied to the control room to provide filtered air which will pressurize the control room. Since the efficiency of the CREFS filters is 95% for elemental/organic iodine and 99% for particulate iodine, the activity is entering the control room via the ventilation system at an estimated rate of 250 cfm for elemental/organic iodine and 50 cfm for particulate iodine. The current licensing basis control room habitability analysis described in FSAR 14.3.5 assumes that when CFEFS is in Mode 4, the unfiltered in-leakage is 10 cfm.

The LOCA assumes that CREFS is actuated by a high radiation signal at the onset of the accident due to the magnitude and timing of the release from containment since the recirculation of the containment sump does not occur until 20 minutes post-accident. The delay in the start of sump recirculation is also true for the RVH drop but automatic actuation of CREFS as a result of containment leakage will not occur. However, there is sufficient time between accident recognition and release initiation to credit manual actuation of CREFS, in addition to, the magnitude of the activity released to the environment is large enough to ensure CREFS actuation via a high radiation signal. Therefore, no delay in CREFS actuation is taken into consideration for the RVH drop control room dose assessment. The assumption that CREFS is operating simultaneously at the initiation of recirculation of the containment sump is maintained for the RVH drop control room dose assessment.

The LOCA ECCS leakage pathway is assumed to contain only elemental iodine. Therefore, under the current licensing basis analysis, the dose to the operator from radioactivity internal to the control room via the ECCS leakage path is primarily due to the elemental iodine activity delivered through the ventilation system. Recent tests of unfiltered inleakage to the control room, while in mode 4, determined that unfiltered inleakage is approximately 100 cfm (Reference 11). Incorporation of the measured unfiltered inleakage value into the RVH drop dose assessment results in an increase that is proportional to the ratio of x/y, where x is the rate of activity delivered to the control room including measured unfiltered inleakage (250 cfm + 100 cfm) and y is the rate of activity delivered under current licensing basis assumptions for unfiltered inleakage (250 cfm + 10 cfm). Therefore, the elemental iodine dose increases by a factor of 1.35 (350 cfm / 260 cfm). Since the ECCS leakage pathway is assumed to contain only elemental iodine activity, the control room doses increase at most by a factor of 1.35.

The offsite RVH ECCS leakage doses are conservatively estimated by dividing the LOCA ECCS leakage dose values by the I-131 scaling factor. The control room external and internal cloud RVH ECCS leakage doses are conservatively estimated by dividing the LOCA ECCS leakage dose values by the I-131 scaling factor then multiplying by 2.7, accounting for the ECCS leakage rate multiplier and the measured unfiltered inleakage. Including the impact of the measured unfiltered inleakage on the external cloud whole body dose is conservative, since this assumption does not actually impact the external cloud dose.

Table 14.3.6-3 below provides the LOCA dose values and the scaled dose values for the RVH drop event. The LOCA ECCS leakage doses are taken from FSAR Table 14.3.5-6, except for the "control room (external)" values. The external cloud control room dose is discussed in FSAR Chapter 11.6 and represents the contribution to the control room operator whole body dose due to the activity external to the control room from release pathways, containment and

ECCS leakage. FSAR 11.6 Reference 2 provides the pathway breakdown of the external cloud dose and was used to determine the ECCS leakage contribution to the external cloud dose. Per Reference 2 of FSAR 11.6, the dose value at 5 feet from the shielded control room window is 0.20 rem and at 10 feet from the shielded control room window is 0.12 rem. The value for control room (external) provided in Table 14.3.6-3 was calculated by summing the location doses (5 feet and 10 feet from shielded control room window) weighted by the time spent in front of the window (25% occupancy at 5 feet and 75% occupancy at 10 feet). Since these values are derived for the design basis LOCA, the accident duration is taken to be 30 days.

Conclusion

As seen in Table 14.3.6-3, the site boundary and low population zone RVH ECCS leakage thyroid and whole body doses are less than the acceptance criteria of 75 rem thyroid and 6.3 rem whole body. The control room doses are less than the acceptance criteria of 30 rem thyroid and 5 rem whole body. Therefore, the bounding dose consequence of the RVH drop can be considered "well within" the 10 CFR 100 limits. The current licensing basis radiological consequence of a design basis LOCA bounds the dose consequence of the postulated RVH drop, and the LOCA remains the limiting event for control room habitability. The source term assumed for the RVH drop event provides a significant margin of safety.

14.3.6.8 References

1. NRC Safety Evaluation, Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendment Re: Incorporation of Reactor Vessel Head Drop Accident Analysis Into the Final Safety Analysis Report," September 23, 2005.

2. NMC Letter from D. L. Koehl to NRC, "Request for Review of Heavy Load Analysis," NRC 2005-0094, July 24, 2005.

3. NRC Safety Evaluation dated June 24, 2005, Point Beach Nuclear Plant, Unit 2 - Issuance of Amendment Re: Incorporation of Reactor Vessel Head Drop Accident Analysis Into the Final Safety Analysis Report," and revised by letter dated August 11, 2005 "Point Beach Nuclear Plant, Unit 2 - Revision to Safety Evaluation for Amendment No. 225."

4. NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," July 2003.

5. NRC Safety Evaluation, Point Beach Nuclear Plant, Unit 2 - Issuance Of Amendment Re: Incorporation Of Reactor Vessel Head Drop Accident Analysis Into The Final Safety Analysis Report, June 24, 2005.

6. NRC Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 3.

7. Sargent & Lundy Calculation 2005-06760, Rev. 3, "Analysis of Postulated Reactor Head Load Drop Onto the Reactor Vessel Flange," July 22, 2005.

8. Westinghouse Calculation Note CN-CRDA-05-68 Rev. 2, "Plastic Analysis of Point Beach Reactor Cooling Piping for Reactor Vessel Head Drop," July 21, 2005.

9. NRC Safety Evaluation dated April 2, 2004, "Point Beach Nuclear Plant, Units 1 And 2 - Issuance Of Amendments Re: Technical Specification 3.9.3, Containment Penetrations, Associated With Handling Of Irradiated Fuel Assemblies And Use Of Selective Implementation Of The Alternative Source Term For Fuel Handling Accident."

10. NRC Regulatory Guide 1.195, "Methods And Assumptions for Evaluating Radiological Consequences Design Basis Accidents At Light-Water Nuclear Power Reactors," May 2003.

11. NMC Letter from A.J. Cayia to NRC dated December 5, 2003 (NRC 2003-0116), "Generic Letter 2003-01, Control Room Habitability - Response To Commitments."

12. Calculation 2005-0033, Rev. 0, "Dose Consequences Post-Reactor Vessel Head Drop at Various Times Post-Shutdown."

Table 14.3.6-1
LOCA and RHV Sump Source Terms

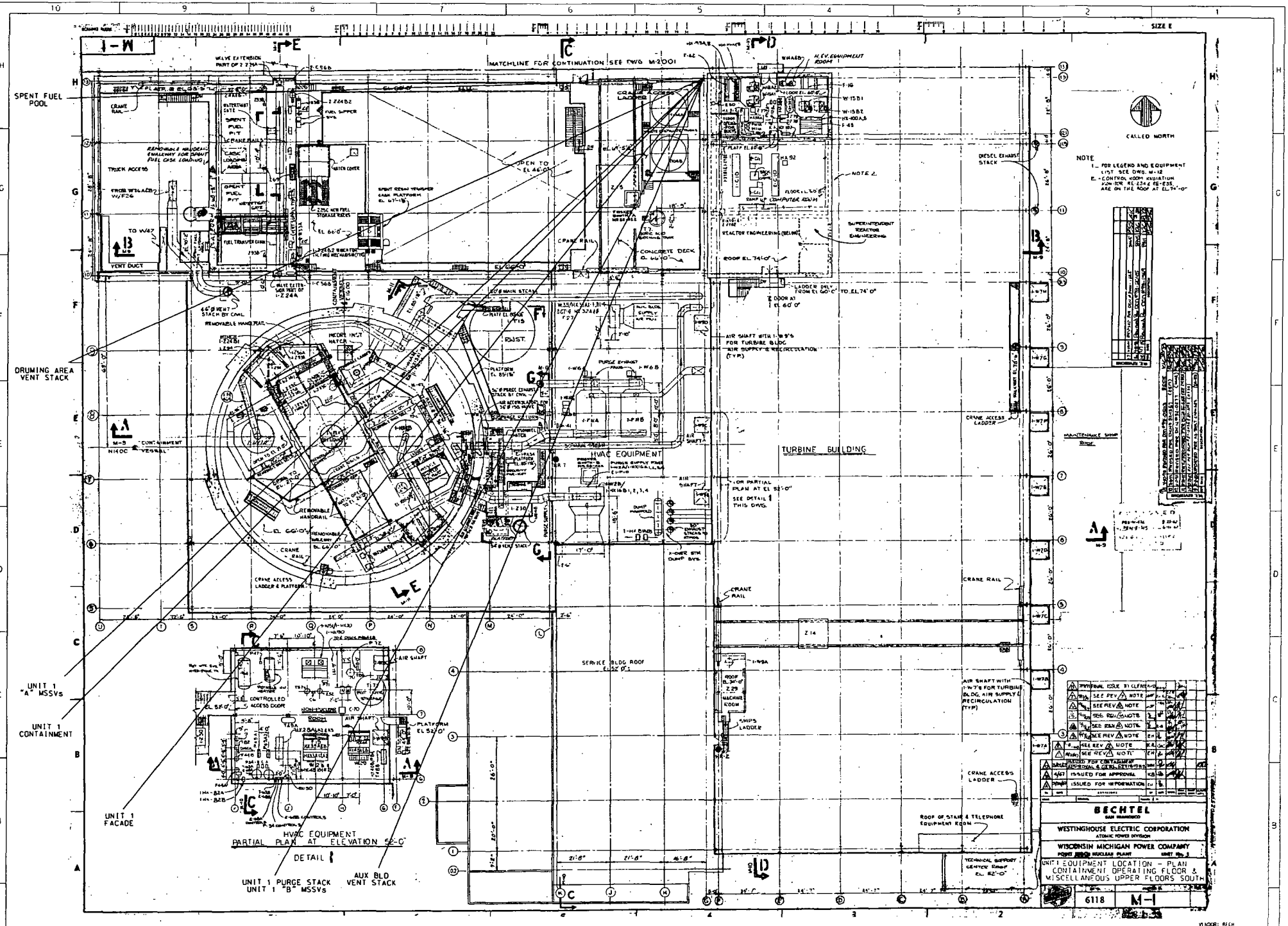
Nuclide	FSAR 14.3.5 LOCA Sump Source Term (Ci)	RVH Drop Sump Source Term (Ci)
I-131	2.07E+07	2.36E+06
I-132	2.96E+07	1.99E+06
I-133	4.23E+07	2.46E+05
I-134	4.65E+07	0.00E+00
I-135	3.95E+07	1.77E+02

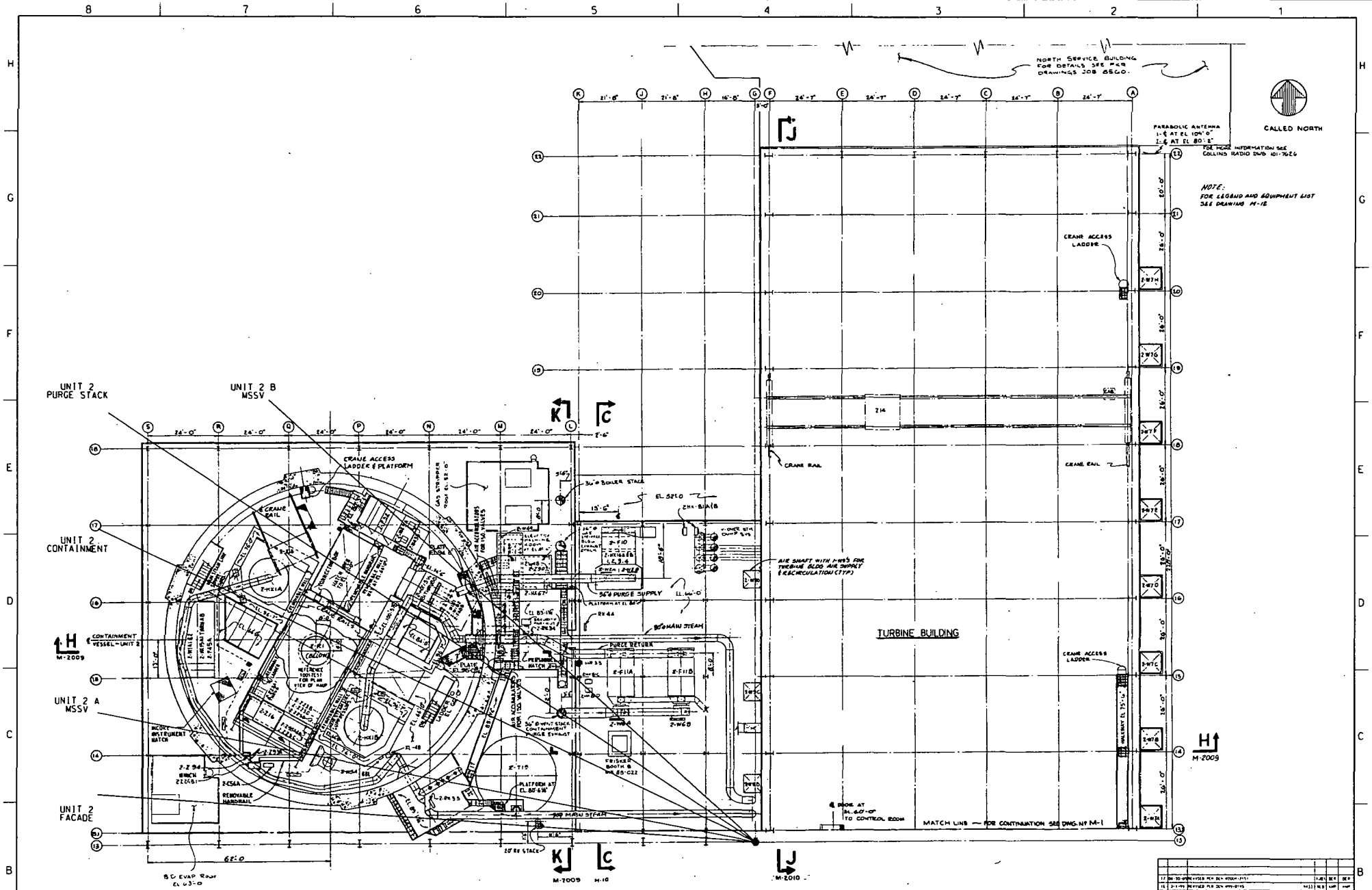
Table 14.3.6-2
Sump Source Term Scaling Factor

Nuclide	Scaling Factor
I-131	8.8
I-132	14.9
I-133	172.0
I-134	-
I-135	2.23E+05

Table 14.3.6-3
RVH Drop Dose Consequence (rem)

	FSAR 14.3.5 LOCA ECCS Leakage (rem)		RVH Drop ECCS Leakage (rem)	
Location	Thyroid	Whole Body	Thyroid	Whole Body
Exclusion Area Boundary	57.12	0.24	5.3	0.022
Low Population Zone	37.0	0.06	3.4	0.006
Control Room	106.7	0.144	26.5	0.04


SIZE E
MATCHLINE FOR CONTINUATION SEE DWG. M-2001
SPENT FUEL POOL
DRUMING AREA VENT STACK
UNIT 1 "A" MSSVs
UNIT 1 CONTAINMENT
UNIT 1 FACADE
UNIT 1 PURGE STACK
UNIT 1 "B" MSSVs
AUX BLD VENT STACK
TURBINE BUILDING
HVAC EQUIPMENT
SERVICE BLDG ROOF EL 52'-0"
CALLED NORTH
NOTE
1. FOR LEGEND AND EQUIPMENT LIST SEE DWG. M-12
2. CONTROL ROOM RADIATION MONITOR RE-234 & RE-235 ARE ON THE ROOF AT EL 74'-0"
HVAC EQUIPMENT PARTIAL PLAN AT ELEVATION 52'-0"
DETAIL 1
BECHTEL
SAN FRANCISCO
WESTINGHOUSE ELECTRIC CORPORATION
ATOMIC POWER DIVISION
WISCONSIN MICHIGAN POWER COMPANY
POINT BEACH NUCLEAR PLANT UNIT No. 2
UNIT 1 EQUIPMENT LOCATION - PLAN
CONTAINMENT OPERATING FLOOR & MISCELLANEOUS UPPER FLOORS SOUTH
6118
M-1
BECH 6118 M-1
EQUIPMENT LOCATION - PLAN
CONTAINMENT OPERATING FLOOR & MISCELLANEOUS UPPER FLOORS SOUTH
POINT BEACH N.P. UNIT 1
35330
18
E
P801889C



THIS DRAWING HAS BEEN SCANNED FROM BECHTEL
DWG. NO. 6118-M2001 SH. 1 REV. 13
DATE 08-15-95 BY LRP/DES ACT NO. 5546

VENDOR : BECH
I.F.I. 15.1
MOD. REQ. :

CGS FILE #8995C

REV	DATE	REVISION DESCRIPTION	DRAWN	CHK	APPROVED
17	06-30-99	REVISED PER DCN #7004-7151	TJB	BEK	BEK
16	3-1-99	REVISED PER DCN #99-8195	MJJ	RLB	JMP
15	04-10-98	REVISED PER DCN #98-8565	BSS	JAC	JRB
14	04-23-98	REVISED & RECODED PER DCN #97-2216	TSM	JMP	JRB
13	06-02-92	REVISED PER DCN #92-2257	TBD	RK	RCW

SCALE 3/32"=1'-0" | WEIGHT | DRAWN F.S. COMPETENTE

BECHTEL
SAN FRANCISCO

WESTINGHOUSE ELECTRIC CORPORATION
ATOMIC POWER DIVISION

POINT BEACH NUCLEAR PLANT
WISCONSIN MICHIGAN POWER CO. UNIT 1
WISCONSIN ELECTRIC POWER CO. UNIT 2

UNIT 2 EQUIPMENT LOCATION PLAN
CONTAINMENT OPERATING FLOOR AND
MISCELLANEOUS UPPER FLOORS NORTH

JOB NO. 6118 | DRAWING NO. M-2001 | REV.

MICROFILM NO 035491 PB 02 MVRL 099 00217

4ASI PB02M0250010 02