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

FPL Energy.
Point Beach Nuclear Plant

December 8, 2008

NRC 2008-0081
10 CFR 50.90

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

Submittal of License Amendment Request 241
Alternative Source Term

Pursuant to 10 CFR 50.90, FPL Energy Point Beach, LLC requests to amend renewed Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant (PBNP), Units 1 and 2, respectively. FPL Energy Point Beach 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, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

FPL Energy Point Beach is requesting approval of a full implementation of the AST. By letter dated April 2, 2004 (ADAMS Accession No. ML040680918), the Nuclear Regulatory Commission (NRC) approved partial implementation of AST for the PBNP fuel handling accident (FHA). This analysis has been revised as discussed in Enclosure 1. To support the AST implementation, FPL Energy Point Beach is also requesting NRC approval of the following:

1. Modification of the control room emergency filtration system (CREFS) to create a new alignment for the accident mode that provides a combination of filtered outside air and filtered recirculation air.

2. Approval to direct continued containment spray (CS) while on sump recirculation, if containment radiological conditions and/or core damage indicates it is required.

3. Use of the Westinghouse RAVE methodology to determine rods in departure from nucleate boiling (DNB) for the locked rotor (LR) event, including the use of other supporting Westinghouse codes (SPNOVA, VIPRE and RETRAN). RAVE is a Westinghouse methodology that has been approved for use by the NRC but has not been approved for this PBNP specific application to determine rods in DNB. Similarly, VIPRE and RETRAN are Westinghouse codes that have been approved for use by the NRC but have not been approved for this PBNP specific application. SPNOVA is part of the ANC code which is currently included in the PBNP licensing basis.

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Enclosure 1 provides a regulatory assessment of the AST implementation including 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 marked up TS Bases pages are also provided in Enclosure 2 for information. Enclosure 3 provides the technical evaluation supporting this LAR. Enclosure 4 provides the Regulatory Issue Summary (RIS) 2006-04 resolution matrix.

Enclosure 5 contains a Westinghouse proprietary report for the RETRAN-SPNOVA-VIPRE (RAVE) methodology, "WCAP-16259 Safety Evaluation Report Compliance." Since PBNP is the first plant to use this methodology, Enclosure 5 (proprietary) of this submittal addresses the conditions and limitations identified in the safety evaluation report (SER) for WCAP-16259-P-A. Also enclosed is Westinghouse authorization letter CAW-08-2479 with accompanying affidavit, Proprietary Information Notice, and Copyright Notice. Since Enclosure 5 contains Westinghouse Electric Company LLC proprietary information, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the Commission may withhold the information from public disclosure and addresses with specificity the considerations listed in Paragraph (b)(4) of Section 2.390 of the Commission's regulations. It is requested that the Westinghouse proprietary information be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations. Correspondence with respect to the copyright or proprietary aspects of the items referenced above or the supporting Westinghouse affidavit should reference CAW-08-2479 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Enclosure 6 contains a non-proprietary version of "WCAP-16259 Safety Evaluation Report Compliance." Enclosure 7 contains the non-proprietary SER Compliance Reports which address the conditions and limitations identified in the SERs for WCAP-10965-P-A (SPNOVA), WCAP-14565-P-A (VIPRE) and WCAP-14882-P-A (RETRAN). Approval of these additional topical reports is requested to support the NRC review and approval of WCAP-16259-P-A for PBNP. SPNOVA is part of the ANC code which is currently included in the PBNP licensing basis.

Enclosure 8 contains a list of new and revised commitments required to implement the AST.

The ARCON96 meteorological data (non-proprietary) and the Westinghouse RADTRAD input decks (proprietary) are being provided on CDs under a separate submittal to facilitate NRC review of the models used in the analysis. This information will be submitted with the LAR, but is under separate cover letter since this is vendor information and cannot be submitted by FPL Energy Point Beach under oath and affirmation. The CD entitled "RADTRAD 3.03 Input Files for Point Beach Alternate Source Term" contains information proprietary to Westinghouse Electric Company LLC, and is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the Commission may withhold the information from public disclosure and addresses with specificity the considerations listed in Paragraph (b)(4) of Section 2.390 of the Commission's regulations. It is requested that the Westinghouse proprietary information be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations.

FPL Energy Point Beach requests approval of the proposed license amendment by February 2010, with the amendment being implemented in the following refueling outage (Unit 1 in the spring of 2010 or Unit 2 refueling outage in the spring of 2011). FPL Energy Point Beach 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 LAR.

FPL Energy 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.

Please address questions regarding this LAR to Mr. Harv Hanneman at (920) 755-7317.

In accordance with 10 CFR 50.91, a copy of this application with the enclosures is being provided to the designated Wisconsin Official.

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

Executed on December 8, 2008

Very truly yours,

FPL Energy Point Beach, LLC

Larry Meyer
Site Vice President

Enclosures (8)

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

ENCLOSURE 1

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

POINT BEACH NUCLEAR PLANT

28 Pages Follow

1.0 SUMMARY DESCRIPTION

1.1 Introduction

Because of advances 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 design basis accident (DBA) radiological consequence analyses. Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating DBA at Nuclear Power Reactors," July 2000 (Reference 8.1), provides guidance on application of Alternative Source Term (AST) in revising the accident source terms used in design basis radiological consequences analyses, as allowed by 10 CFR 50.67.

Pursuant to the requirements of 10 CFR 50.90 and 10 CFR 50.67, FPL Energy Point Beach, LLC proposes to amend Appendix A of Facility Operating Licenses DPR-24 and DPR-27, Technical Specifications (TS) for Point Beach Nuclear Plant (PBNP), Units 1 and 2, respectively. This amendment request incorporates a revision to the PBNP current licensing basis (CLB) by revising the accident source term used in the design basis radiological analyses supporting a full-implementation application of an AST methodology. Proposed TS changes, which are supported using the AST in the DBA radiological consequence analyses, are included in this application for a license amendment as Enclosure 2. The associated TS bases changes are also included for staff information.

Five new commitments for plant modifications and procedure changes are made in support of this application:

- FPL Energy Point Beach will modify the PBNP control room (CR) radiation shielding to ensure CR habitability requirements are maintained. This modification is scheduled to be completed following Nuclear Regulatory Commission (NRC) approval of this LAR 241 "Alternative Source Term" at the Unit 1 (2010) refueling outage.

- FPL Energy Point Beach will modify the containment spray (CS) and residual heat removal (RHR) systems to provide throttling capability of CS and RHR during the emergency core cooling system (ECCS) recirculation phase. These modifications will be completed on a unit specific basis at the next Unit 1 (2010) and Unit 2 (2009) refueling outages.

- FPL Energy Point Beach will revise PBNP Emergency Operating Procedures (EOPs) to direct continued containment spray while on sump recirculation, if containment radiological conditions and/or core damage indicates it is required. These procedure changes will be implemented following NRC approval of this LAR 241 "Alternative Source Term" and following the completion of each unit specific installation of the CS and RHR system modifications to provide throttling capability during the ECCS recirculation phase.

- FPL Energy Point Beach will modify the control room emergency filtration system (CREFS) to create a new alignment for the accident mode that provides a combination of filtered outside air and filtered recirculation air. The modifications will include redundancy for all CREFS active components that must reposition from their normal operating position, and auto-start capability on loss of offsite power in conjunction with a containment isolation or high control room radiation signal from a diesel generator supplied source for the CREFS fans required for the new system alignment. This modification will be completed following NRC approval of this LAR 241 "Alternative Source Term" during the second site refueling outage that completes installation of the CS and RHR system modifications to provide throttling capability during the ECCS recirculation phase, thus completing installation for both units.

- FPL Energy Point Beach will modify the primary auxiliary building (PAB) ventilation system (VNPAB) to ensure redundancy of active components needed to operate the PAB exhaust system. VNPAB components credited for AST will be upgraded to an augmented quality status. No credit is taken by AST for the PAB charcoal filters. FPL Energy Point Beach will revise PBNP EOPs to address starting the VNPAB fans.

The regulatory commitment provided in response to GL 2003-01 dated May 1, 2007 (Adams Accession No. ML071210471) stating "NMC will provide Technical Specification changes to reference an acceptable surveillance methodology (and plans for any associated plant modifications to the CRE) to support requested information in GL 2003-01, Item (c), for PBNP no later than 180 days following NRC approval of TSTF-448." is revised for the AST as follows:

- FPL Energy Point Beach will submit a LAR addressing CR habitability surveillance methodology in accordance with TSTF-448, as modified by TSTF-508, within 60 days of approval of the AST LAR.

The following regulatory commitments will be eliminated:

- As a part of PBNP's response to NUREG-0737, Item III.D.3.4 "Control Room Habitability Requirements," documented in NRC Safety Evaluation Report (SER) dated August 10, 1982 (not available in ADAMS), PBNP committed to the placement of portable lead shielding to limit radiation exposure through the CR doors and windows. Following installation of the permanent shielding modification for the CR that replaces the temporary shielding, this commitment will be eliminated.

- As a part of PBNP's application dated July 24, 2005, incorporating a PBNP reactor vessel head drop (RVHD) accident analysis and subsequent NRC SER dated September 23, 2005 (Adams Accession No. ML052560089) (Reference 8.16), PBNP committed to a reactor shutdown time of 100 hours prior to lifting the reactor vessel head. Following NRC approval of this amendment request for the RVHD radiological analysis using the AST methodology, this commitment will be eliminated, as the proposed RVHD radiological analysis using the AST methodology assumes the reactor head is lifted immediately following reactor shutdown.

The PBNP current licensing basis (CLB) for radiological consequence analyses of accidents discussed in the Final Safety Analysis Report (FSAR) Chapter 14 (sections 14.1.8, 14.2.4, 14.2.5, 14.2.6, 14.3.5 and 14.3.6) are based upon methodologies and assumptions that are primarily derived from Technical Information Document (TID)-14844 (Reference 8.2) and other early guidance, except for the fuel handling accident (FHA). The FHA radiological consequences analysis (FSAR 14.2.1) currently uses the AST methodology as approved previously by the NRC staff on April 4, 2004, by the issuance of Amendments 213 and 218 (Reference 8.3).

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

The Westinghouse RAVE methodology was used to determine the percentage of fuel rods in departure from nucleate boiling (DNB) for the locked rotor (LR) event. RAVE is a methodology that uses inputs from three other Westinghouse codes that have been approved for use by the NRC: RETRAN, ANC (SPNOVA) and VIPRE. Westinghouse WCAP-16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," August 2006 (ADAMS Accession No. ML062580352) (Reference 8.4) was submitted and approved for use by the NRC. Enclosures 5 (proprietary) and 6 (non-proprietary) provide the WCAP-16259-P-A SER Compliance addressing the conditions and limitations identified in the NRC SER that is required to use this methodology. Enclosure 7 provides the SER Compliance Matrices for ANC (SPNOVA), (WCAP-10965-P-A), VIPRE, (WCAP-14565-P-A), and RETRAN, (WCAP-14882-P-A) (References 8.5, 8.6 and 8.7, respectively) and addresses the conditions and limitations identified in their associated SERs. To support the AST implementation, FPL Energy Point Beach is requesting NRC approval to use the Westinghouse methodology RAVE to determine rods in DNB for the LR event, including the use of the three other supporting Westinghouse codes (SPNOVA, VIPRE and RETRAN).

1.2 <u>Evaluation Overview and Objective</u>

As documented in NEI 99-03, "Control Room Habitability Assessment Guidance," Revision 1, March 2003 (Reference 8.8) and NRC Generic Letter (GL) 2003-01 "Control Room Habitability," June 12, 2003 (Reference 8.9), several nuclear plants performed testing on CR unfiltered air inleakage that demonstrated leakage rates in excess of amounts assumed in the current accident analyses. A PBNP tracer gas test demonstrated CR inleakage greater than that assumed in the current FSAR radiological analyses. The AST methodology in RG 1.183, as supplemented by Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," March 7, 2006 (Reference 8.10), is being used in this application to calculate the offsite and CR radiological consequences for PBNP to support the CR habitability program by establishing a conforming set of radiological analyses for the accidents listed below.

The following FSAR Chapter 14 accidents are analyzed:

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

The FHA is currently approved for use with the AST methodology (Reference 8.3). The revised FHA dose analysis presented in this license amendment request (LAR) determined that the exclusion area boundary (EAB), low population zone (LPZ), and the CR doses meet RG 1.183 acceptance criteria. Input assumption changes are discussed in Enclosure 3, Section 6.6 of this LAR.

The LR event is being analyzed using the RAVE methodology described in WCAP16259-P-A to determine rods in DNB. RAVE is a methodology that uses inputs from three other Westinghouse codes that have been approved for use by the NRC: RETRAN, ANC (SPNOVA) and VIPRE. Enclosures 5 (proprietary) and 6 (non-proprietary) provide the WCAP-16259-P-A SER Compliance addressing the conditions and limitations identified in the NRC SER that is required to use this methodology. Enclosure 7 provides the SER Compliance Matrices for ANC (SPNOVA) (WCAP-10965-P-A), VIPRE (WCAP-14565-P-A), and RETRAN (WCAP-14882-P-A) and addresses the conditions and limitations identified in the associated SERs.

2.0 PROPOSED CHANGES

2.1 Licensing Bases Changes

FPL Energy Point Beach proposes to revise the PBNP CLB 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, described in Enclosure 3 with more detail provided, are incorporated in the analyses:

- 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 CR intake are re-analyzed for existing pathways using NUREG/CR-6331, "ARCON96; Atmospheric Relative Concentrations in Building Wakes," May 1997 (Reference 8.12).
- New values for CR unfiltered air inleakage assumed to bound the test results, are modeled.
- The CR ventilation system will operate with filtered recirculation, in addition to filtered fresh air intake. (This includes auto-loading of the CREFS fans onto their associated emergency diesel generator during a loss of offsite power (LOOP) coincident with a LOCA.)
- Credit is taken for future shielding modifications to the CR.
- Reduced values for the allowable dose equivalent (DE) I-131 concentrations in the primary and secondary systems are used.
- A reduced allowable containment leakage is modeled.
- A factor of two increase over FPL Energy Point Beach's commitment to Item III.D.1.1 of NUREG-0737 is applied to the ECCS leakage limit for CR habitability radiological analyses as discussed below.
  - ECCS recirculation leakage into the PAB is assumed to be 300 cc/min.
  - ECCS back-leakage to the refueling water storage tank (RWST) is assumed to be 500 cc/min.
- Credit is taken for CS removal of fission products while on ECCS recirculation (LOCA).
- Flashing fractions are applied to the SGTR break flow.
- The need for post-accident reliance on potassium iodide (KI) for CR operators is removed from the PBNP licensing basis.

- The requirement that the reactor must be shut down for 100 hours prior to lifting the reactor vessel head is eliminated based on the revised RVHD analysis.

- Revised radial core power peaking factors are applied to the radiological analyses.

- FHA gap fractions have been increased to reflect the fact that some nuclear fuel assemblies exceed the RG 1.183, Table 3, Footnote 11 criteria. As a conservative approach, the gap fractions are those from RG 1.25 (Reference 8.13) with the value for I-131 increased by 20%, consistent with the recommendation in NUREG/CR-5009 (Reference 8.14).

- For the CRDE, the primary to secondary leak rate duration was determined based on time required for the primary system pressure to decrease to the secondary system pressure. The primary system pressure decrease was based on a small-break LOCA (about 2" diameter). The depressurization time has been conservatively increased by 25% above the calculated depressurization time.

- For the LR event, 30% of the fuel rods in the core are assumed to suffer damage due to DNB. The RAVE methodology "Westinghouse WCAP-16259-P-A" is being incorporated into the CLB for the LR event to determine rods in DNB. RAVE is a methodology that uses inputs from three other Westinghouse codes that have been approved for use by the NRC: RETRAN, ANC (SPNOVA) and VIPRE.

- Credit is taken for manual operator action to restore PAB ventilation (VNPAB) within 30 minutes following the alignment of RHR to containment sump recirculation mode of operation. If a LOCA occurs coincident with a LOOP, the VNPAB will be manually restarted to ensure that the auxiliary building vent stack is the source of the release associated with the ECCS leakage phase of the event. This configuration is an input assumption for the LOCA radiological analysis.

A table providing a comparison of the input assumptions for the CLB, the withdrawn 2007 AST submittal, and this submittal for each analyzed accident analyzed is provided in Appendix A to Enclosure 3.

2.2 Technical Specification Changes

The following changes to the PBNP TS are proposed. (Enclosure 2 contains marked-up TS and TS Bases pages):

- 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.

- TS Section 3.4.16, RCS Specific Activity

  Surveillance Requirement (SR) 3.4.16.2 is revised to change the specific activity of the reactor coolant from DE I-131 ≤ 0.8 μCi/gm to ≤ 0.5 μCi/gm.

  Note that the current TS Condition C for DE I-131 > 50 μCi/gm is not revised. The AST radiological analyses use a value of 60 μCi/gm which is conservative.

- TS Section 3.7.9, Control Room Emergency Filtration System (CREFS)

  SR 3.7.9.3 is revised to delete the word "makeup" which changes the SR to read "Verify each CREFS emergency fan actuates on an actual or simulated actuation signal."

  SR 3.7.9.6 is revised to delete the word "makeup" twice which changes the SR to read "Verify each CREFS emergency fan can maintain a positive pressure of ≥ 0.125 inches water gauge in the CR envelope, relative to the adjacent turbine building during the emergency mode of operation at a flow rate of 4950 cfm ±10%."

- TS Section 3.7.13, Secondary Specific Activity

  LCO 3.7.13 is revised to change the specific activity of the secondary coolant from ≤ 1.00 μCi/gm to ≤ 0.1 μCi/gm DE I-131.

  SR 3.7.13.1 is revised to change the specific activity of the secondary coolant from ≤ 1.00 μCi/gm to ≤ 0.1 μCi/gm DE I-131.

- TS Section 5.5.15, Containment Leakage Rate Testing Program

  Item c. is revised to change the maximum allowable containment leakage rate, L_a at P_a from 0.4% to 0.2% of containment air weight per day.

- 5.6.4 CORE OPERATING LIMITS REPORT (COLR)

  The list entitled, "The approved analytical methods are described in the following documents: " is revised to include a new item (13).

  (13) WCAP-16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Analyses"

The information provided in the TS Basis 3.7.9 for Control Room Ventilation (VNCR) Modes 1, 2, 3, and 4 will be deleted from the PBNP TS Basis 3.7.9 as this information does not directly describe the DBA function of the system, and is duplicative of the information that is currently provided in the PBNP FSAR Section 9.8.

2.3 Credited Plant Modifications and Procedure Changes (proposed commitments)

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. To support specific assumptions used in the AST radiological consequence analyses, plant modifications are necessary. The following modifications are required for implementation of the AST and are new commitments.

- FPL Energy Point Beach will modify the PBNP CR radiation shielding to ensure CR habitability requirements are maintained. This modification will be installed under 10 CFR 50.59 and is scheduled to be completed following NRC approval of LAR 241 "Alternative Source Term" at the Unit 1 (2010) refueling outage.

  This modification to the CR shielding will install permanent shielding that replaces the temporary shielding installed in response to NUREG-0737 Item III.D.3.4. Current radiological analyses assume placement of 0.5 inches of lead-equivalent thickness shielding for the window and 0.25 inches of lead-equivalent thickness shielding for the doors to reduce the post-accident dose from the passing plume. The temporary shielding is being replaced with permanent shielding and additional shielding is being installed around the CRE. The shine doses to the CR operators will be reduced by shielding modifications made to the CR outside of the CRE.

  As a part of PBNP's response to NUREG-0737, Item III.D.3.4 "Control Room Habitability Requirements," documented in NRC SER dated August 10, 1982 (not available in ADAMS) (Reference 8.17), PBNP committed to the placement of portable lead shielding to limit radiation exposure through the CR doors and windows. Following installation of the permanent shielding modification for the CR that replaces the temporary shielding, this commitment will be eliminated.

- FPL Energy Point Beach will modify CREFS to create a new alignment for the accident mode that provides a combination of filtered outside air and filtered recirculation air. The modifications will include redundancy for all CREFS active components and auto-start capability on loss of offsite power from a diesel generator supplied source for the CREFS fans required for the new system alignment. FPL Energy Point Beach is requesting NRC approval of this new operational mode.

  A new operational mode for CREFS, known as Mode 5 will be established. This mode is referred to as VNCR accident mode to avoid confusion with plant operating MODES in the TS. This change will provide for a combination of filtered outside air and filtered recirculation. The VNCR accident mode will provide a total flow rate of 4950 cfm ±10% with a minimum of 1955 cfm of filtered return air. Tracer gas testing in the proposed VNCR accident mode, which is a combination of current VNCR operational Modes 3 and 4, has been performed with satisfactory results.

- FPL Energy Point Beach will modify the CS and RHR systems to provide throttling capability of CS and RHR during the ECCS recirculation phase. CS on recirculation following a LOCA is an assumption in the AST dose calculations. These modifications will be installed under 10 CFR 50.59 and will be completed on a unit specific basis at the next Unit 1 (2010) and Unit 2 (2009) refueling outages.

  The flow path for the modification to the CS and RHR systems to provide throttling capability during the ECCS recirculation phase is provided in Enclosure 3, Figure 3.

The AST LOCA dose analysis assumes CS is operated for three 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 low head safety injection flow is required to maintain adequate RHR pump net positive suction head (NPSH) and to meet the flow requirements for Generic Safety Issue (GSI)-191, "Assessment Of Debris Accumulation On PWR Sump Performance" (Reference 8.15). Current FSAR radiological accident analyses do not take credit for operation of the CS system in the ECCS recirculation phase.

- FPL Energy Point Beach will revise PBNP EOPs to direct continued CS while on sump recirculation, if containment radiological conditions and/or core damage indicates it is required. FPL Energy Point Beach is requesting NRC approval of this change. These procedure changes will be implemented following NRC approval of LAR 241 "Alternative Source Term" and following the completion of each unit specific installation of the CS and RHR system modifications to provide throttling capability during the ECCS recirculation phase.

The dose calculations prepared in support of this submittal assume that CS 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 postulated damaged core.

3.0 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," contains offsite dose limits in terms of whole body and thyroid dose, and references TID-14844 (Reference 8.2).

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 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 operating power reactor licensees 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 analysis assumptions are described in Enclosure 3 and summarized in Section 4.0 of this enclosure. The analysis assumptions are consistent with RG 1.183 except as listed below:

- FHA gap fractions have been increased to reflect the fact that some nuclear fuel assemblies exceed the RG 1.183, Table 3, Footnote 11 criteria. As a conservative approach, the gap fractions are those from RG 1.25 (Reference 8.13) with the value for I-131 increased by 20%, consistent with the recommendation in NUREG/CR-5009 (Reference 8.14).

- For the CRDE, the primary to secondary leak rate duration was determined based on time required for the primary system pressure to decrease to the secondary system pressure. The primary system pressure decrease was based on a small-break LOCA (about 2" diameter). The depressurization time has been conservatively increased by 25% above the calculated depressurization time.

- For a LOCA, manual operator actions are required to align the CS and RHR systems for CS on recirculation from the containment sump.

- The VNCR system will operate with filtered recirculation in addition to filtered outside air intake. The VNCR system is classified as non-safety related, however selected components such as the CR charcoal filter fans, the CR recirculation fans, damper controllers and the charcoal/HEPA/roughing filter have been upgraded to augmented quality status. For the accident mode of operation, the two radiation monitor actuation signals are diverse and the radiation monitors are augmented quality status. The containment isolation actuation signals are safety-related and redundant. The fans, dampers and filter are tested in accordance with Surveillance Requirements SR 3.7.9.1 through SR 3.7.9.6 of the Point Beach Technical Specifications (see page 3.7.9-2 of Enclosure 2). In order to address the requirements of Section 5.1.2 of Regulatory Guide 1.183, the system will be modified to ensure redundancy of active components and to auto-start the system from the safety related diesel generators on loss of offsite power.

- Credit is taken for manual operator action to restore the VNPAB within 30 minutes following the alignment of RHR to containment sump recirculation mode of operation. If a LOCA occurs coincident with a LOOP, the VNPAB will be manually restarted to ensure that the auxiliary building vent stack is the source of the release associated with the ECCS leakage phase of the event. This configuration is an input assumption for the LOCA radiological analysis.

The criteria of 10 CFR 50.67 were used to evaluate the PBNP CLB DBAs for radiological consequences.

3.1 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 GDCs to which the plant was licensed. The PBNP GDCs are similar in content to the draft GDCs 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 11, Control Room, states that the facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the control room under any credible post-accident condition or as an alternative, access to other areas of the facility as necessary to shut down and maintain safe control of the facility without excessive radiation exposures to personnel.

  The proposed changes remove the use of KI from the PBNP licensing basis, provide additional shielding for the CR, add a new VNCR accident mode, and provide updated calculations for CR doses. The revised dose calculations demonstrate that the dose to the CR operators is less than 5.0 Rem acceptance criteria for all of the accidents addressed in this submittal. Thus, the proposed changes provide additional assurance that the requirements of PBNP GDC 11 are met.

- 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 meeting the requirements of PBNP GDC 70, which addresses radioactive waste systems. The volume control tank (VCT) and waste gas decay tank rupture events have been relocated from FSAR Chapter 14 to Chapter 11 (Section 11.1, "Liquid Waste Management System" and Section 11.2, "Gaseous Waste Management System"). These two events are not analyzed for AST. However, the events meet the acceptance criteria of 10 CFR 100. 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.

- TS 5.6.4 Core Operating Limits Report (COLR) approved analytical methods is revised to include WCAP-16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Analyses." RAVE is a methodology that uses inputs from three other Westinghouse codes that have been approved for use by the NRC: RETRAN, ANC (SPNOVA) and VIPRE.

Five new commitments for plant modifications and procedure changes are made in support of this application:

- FPL Energy Point Beach will modify the PBNP CR radiation shielding to ensure CR habitability requirements are maintained. This modification is scheduled to be completed following NRC approval of LAR 241 "Alternative Source Term" at the Unit 1 (2010) refueling outage.

- FPL Energy Point Beach will modify the CS and RHR systems to provide throttling capability of CS and RHR during the ECCS recirculation phase. These modifications will be completed on a unit specific basis at the next Unit 1 (2010) and Unit 2 (2009) refueling outages.

- FPL Energy Point Beach will revise PBNP EOPs to direct continued CS while on sump recirculation, if containment radiological conditions and/or core damage indicates it is required. These procedure changes will be implemented following NRC approval of LAR 241 "Alternative Source Term" and following the completion of each unit specific installation of the CS and RHR system modifications to provide throttling capability during the ECCS recirculation phase.

- FPL Energy Point Beach will modify CREFS to create a new alignment for the accident mode that provides a combination of filtered outside air and filtered recirculation air. The modifications will include redundancy for all CREFS active components and auto-start capability on loss of offsite power from a diesel generator supplied source for the CREFS fans required for the new system alignment. This modification will be completed following NRC approval of LAR 241 "Alternative Source Term" during the second site refueling outage that completes installation of the CS and RHR system modifications to provide throttling capability during the ECCS recirculation phase, thus completing installation for both units.

- FPL Energy Point Beach will modify the VNPAB to ensure redundancy of active components needed to operate the PAB exhaust system. VNPAB components credited for AST will be upgraded to an augmented quality status. No credit is taken by AST for the PAB charcoal filters. FPL Energy Point Beach will revise PBNP EOPs to address starting the VNPAB fans.

The regulatory commitment provided in response to GL 2003-01 dated May 1, 2007 (Adams Accession No. ML071210471) stating "NMC will provide Technical Specification changes to reference an acceptable surveillance methodology (and plans for any associated plant modifications to the CRE) to support requested information in GL 2003-01, Item (c), for PBNP no later than 180 days following NRC approval of TSTF-448." is revised for the AST as follows:

- FPL Energy Point Beach will submit a LAR addressing CR habitability surveillance methodology in accordance with TSTF-448, as modified by TSTF-508, within 60 days of approval of the AST LAR.

The following regulatory commitments will be eliminated:

- As a part of PBNP's response to NUREG-0737, Item III.D.3.4 "Control Room Habitability Requirements," documented in NRC SER dated August 10, 1982 (not available in ADAMS), PBNP committed to the placement of portable lead shielding to limit radiation exposure through the CR doors and windows. Following installation of the permanent shielding modification for the CR that replaces the temporary shielding, this commitment will be eliminated.

- As a part of PBNP's application dated July 24, 2005, incorporating a PBNP RVHD accident analysis and subsequent NRC SER dated September 23, 2005 (ML052560089) (Reference 8.16), PBNP committed to a reactor shutdown time of 100 hours prior to lifting the reactor vessel head. Following NRC approval of this amendment request for the RVHD radiological analysis using the AST methodology, this commitment will be eliminated, as the proposed RVHD radiological analysis using the AST methodology assumes the reactor head is lifted immediately following reactor shutdown.

4.0 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.

4.1 Radiological Evaluation Methodology

- Acceptance Criteria

  Offsite and CR 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 RG 1.183, Table 6. 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 VNCR and dose calculation model, radiation source terms, and χ/Q factors. Event-specific assumptions are discussed in the event analyses contained in Enclosure 3.

  The Westinghouse RAVE methodology was used to determine rods in DNB for the LR event. The topical report for RAVE (Westinghouse WCAP-16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," August 2006 (ADAMS Accession No. ML062580352) (Reference 8.4)) was submitted and approved for use by the NRC. RAVE is a methodology that uses inputs from three other Westinghouse codes that have been approved for use by the NRC: ANC (SPNOVA), (WCAP-10965-P-A), VIPRE, (WCAP-14565-P-A), and RETRAN, (WCAP-14882-P-A) (References 8.5, 8.6 and 8.7, respectively). RAVE, VIPRE and RETRAN are Westinghouse codes that have been approved for use by the NRC but have not been approved for this PBNP specific application to determine rods in DNB. SPNOVA is part of the ANC code which is currently included in the PBNP licensing basis.

  To support the AST implementation, FPL Energy Point Beach is requesting NRC approval to use the Westinghouse RAVE methodology to determine rods in DNB for the LR event, including the use of the three other supporting Westinghouse codes (ANC (SPNOVA), VIPRE and RETRAN).

4.2 Fission Product Inventory and Source Terms

- 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 1800 MWt with 0.6% uncertainties (analyzed core power of 1811 MWt), and uprated values of fuel enrichment and burnup. The core nuclide inventory also includes a multiplier of 1.04, adding margin to the core inventory, to account for cycle-to-cycle variations in enrichment, cycle burnup, and loading that may not have been explicitly addressed in the parametric ranges included in the core inventory modeling using the ORIGEN-S computer code.

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 Section 3.0 provides additional information on the new core source term.

- Reactor Coolant Source Term - For the Reactor Coolant System (RCS), 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 rods producing 1% of the power containing fuel defects operating at an analyzed core power of 1811 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 Section 3.0. The activities in analyses are adjusted to TS limits for DE I-131 and Xe-133.

- Gap Inventory for Non-LOCA Accidents - The gap fractions listed in Section 3.2 of RG 1.183 (Reference 8.1) 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. With the exception of the FHA damaged fuel, no other failed fuel is assumed to exceed the criteria of RG 1.183, Table 3, Footnote 11.

  The FHA is currently approved to use the AST methodology. For the FHA, higher gap fractions were applied assuming the fuel does not meet the limits on burnup and linear heat generation rate, following the method approved by the NRC for Kewaunee (Reference 8.20) (ML070430020). The gap fractions assumed are those from RG 1.25 (Reference 8.13) with the value for I-131 adjusted consistent with the recommendation of NUREG/CR-5009 (Reference 8.14). The detailed gap inventory for non-LOCA accidents is addressed in Section 3.3 of Enclosure 3.

  For the LR event, 30% of the fuel rods in the core are assumed to suffer damage due to DNB, as a result of the locked rotor, sufficient that all of their gap activity is released to the RCS. The RAVE methodology "Westinghouse WCAP-16259-P-A" is being incorporated into the CLB for the LR event to determine rods in DNB. The rods failing in DNB are high power first or second cycle rods which meet the burnup criteria of RG 1.183, Table 3, Footnote 11.

4.3 Atmospheric Relative Concentrations

The χ/Q values for the PBNP EAB and the 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.

The CR intake χ/Q values are based on meteorological data collected at PBNP from September 2000 to September 2005. The CR intake χ/Q values are calculated using "ARCON96: Atmospheric Relative Concentrations in Building Wakes" methodology (Reference 8.12). For additional information, refer to Enclosure 3, Section 4.0. The ARCON96 meteorological data are being provided on CD in a separate submittal to facilitate NRC review of the models used in the analysis.

4.4 Current Licensed Power Level and AST Modeling

- Current Licensed Power Level

  The current licensed maximum reactor core power level is 1540 MWt. The analyses in this enclosure model a maximum core power of 1800 MWt with 0.6% uncertainties (analyzed core power of 1811 MWt). This power level was chosen to support a future extended power uprate (EPU) LAR and is conservative with respect to the current licensed power level. Although the analyses were performed at a higher power level, this LAR is not requesting approval for use of the higher core power level.

  The FHA, based on the AST, was submitted previously under Reference 8.11 and approved under Reference 8.3. The previously approved analysis, performed at 1650 MWt, bounds the current licensed power level of 1540 MWt. The revised FHA dose analysis presented in this LAR performed at an analyzed core power of 1811 MWt is intended to support a future EPU LAR and is conservative with respect to the current licensed power level for the FHA.

- Methodologies

  The AST analyses performed for PBNP use assumptions and models defined in RG 1.183 to provide appropriate and prudent safety margins. Deviations from RG 1.183 are clearly defined. The methodologies used for source terms and dose calculations are NRC-sponsored and industry-standards computer codes. The RAVE methodology "Westinghouse WCAP-16259-P-A" is being incorporated into the CLB for the LR event to determine rods in DNB. RAVE is a methodology that uses inputs from three other Westinghouse codes that have been approved for use by the NRC: ANC (SPNOVA), VIPRE, and RETRAN. 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, conservative assumptions have been used in the dose analyses to develop source terms, and identify release pathways. A discussion of the assumptions is provided for each event in Enclosure 3, Section 6.0.

4.5 PBNP Design Basis Accidents

This section provides a brief description of the accidents addressed in the AST. Detailed discussion of these accidents is provided in Enclosure 3, Section 6.0.

LOCA

Radiological consequences due to a LOCA are due to a postulated abrupt failure of the main reactor coolant piping. Activity from the core is released to the containment and from there, released to the environment by containment leakage and leakage from the ECCS.

Non-LOCA Accidents

The contribution of shine is not explicitly determined for non-LOCA accidents. There is sufficient margin for the doses calculated for the non-LOCA accidents that, assuming the worst case shine (LOCA), the non-LOCA accidents will not exceed the applicable acceptance criteria.

- 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 and primary-to-secondary leakage 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 (SG) tube is assumed to occur. At the start of the accident, radionuclides from the primary coolant enter the SG, 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 prior to reactor trip. The primary-to-secondary break flow results in depressurization of the RCS. Reactor trip and safety injection (SI) are assumed to be automatically initiated simultaneously on low pressurizer pressure. For calculating dose rates, 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 SG safety or atmospheric dump valves (ADVs).

  Following reactor trip and SI actuation, 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 the operators have completed the actions necessary to terminate the steam release from the ruptured SG. Pressure between the ruptured SG and the primary system is such that the ruptured SG is not overfilled.

- 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 is predicted to occur due to DNB as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed SG tube leakage, fission products are discharged from the primary system into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the ADVs or main steam safety valves (MSSVs). In addition, iodine and alkali metal 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 SGs following the accident.

- MSLB

The complete severance of a main steam line outside containment is assumed to occur. The affected SG rapidly depressurizes and releases radioiodines and alkali metals initially contained in the secondary coolant. In addition, primary coolant activity is transferred via SG tube leaks to the outside atmosphere. A portion of the iodine and alkali metal activity, initially contained in the intact SG, and activity due to tube leakage, is released to the atmosphere through either the ADVs or the MSSVs. For radiological consequences analyses, the steam line break outside containment bounds any steam line break inside containment, since the outside break provides a means for direct release to the environment.

- CRDE

A mechanical failure of control rod mechanism pressure housing is assumed to have 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 damage are assumed to occur. Due to the pressure differential between the primary system 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 ADVs or the MSSVs. Iodine and alkali metal 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 SGs 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.

- FHA

The FHA currently uses the AST methodology as approved previously by the NRC staff on April 4, 2004, by issuance of amendments 213 and 218. The revised FHA dose analysis presented in this LAR indicates that the EAB, LPZ, and CR doses meet RG 1.183 acceptance criteria.

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 at the spent fuel pool.

- 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 100% 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 SI lines is not predicted.

4.6 Summary of Results

Results of the PBNP radiological consequence analyses using the AST methodology and the corresponding allowable CR unfiltered air inleakage have determined that the doses of the CR, LPZ and EAB do not exceed the requirements of 10 CFR 50.67. The results are summarized in the table in Enclosure 3, Section 1.2. Enclosure 3, Section 6.0 explains these results and acceptance criteria in more detail. Enclosure 3 supports a maximum allowable CR unfiltered air inleakage of 200 cfm for LOCA and MSLB, and 300 cfm for the remaining non-LOCA events.

4.7 Conclusion

The offsite and CR doses, for the radiological accidents described in Chapter 14 of the PBNP FSAR, have been reanalyzed, consistent with the AST methodology described in RG 1.183. The calculated doses for the EAB, LPZ and CR are all within the acceptance criteria of 10 CFR 50.67. Post-accident administration of KI to CR personnel is removed from the PBNP licensing basis.

5.0 REGULATORY SAFETY ANALYSIS

5.1 General Design Criteria

PBNP was licensed prior to the 1971 publication of 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants." As such, PBNP is not licensed to the Appendix A GDCs. PBNP Updated FSAR, Section 1.3, lists the plant-specific GDCs to which the plant was licensed. The PBNP 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 RGs and criteria. Use of the new analysis method replaces 10 CFR 100 as the applicable dose acceptance criteria for the DBAs identified in Section 1 of this enclosure.

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 CR occupants shall not exceed 5 rem TEDE dose. The analysis provided to support the requested changes demonstrates that this requirement is met.

FPL Energy Point Beach 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, FPL Energy Point Beach will update the FSAR to reflect the proposed new analysis method. The changes to the TSs incorporate assumptions used in the new analysis.

Compliance with PBNP GDCs 11, 17, 18, and 70 is described in Section 4.

5.2 FSAR Accident Analysis Compliance

FPL Energy Point Beach 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 FSAR Chapter 14 accidents:

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

As part of the full implementation of this AST, the changes assumed in the analysis as part of full implementation of the AST are described in Section 2.0 of this enclosure.

5.3 Compliance with RG 1.183, "Alternative Radiological Source Terms for Evaluating DBA at Nuclear Power Reactors"

Enclosure 3 contains the technical evaluation which incorporates the guidance of RG 1.183. Enclosures 5 (proprietary) and 6 (non-proprietary) provide the Westinghouse WCAP-16259-P-A SER Compliance required to use the RAVE methodology. Enclosure 7 provides the SER Compliance Matrices for ANC (SPNOVA), (WCAP-10965-P-A), VIPRE, (WCAP-14565-P-A), and RETRAN, (WCAP-14882-P-A)

5.4 Environmental Qualification (EQ)

Section 1.3.5 of the RG 1.183 discusses the position on performance of required EQ analyses with respect to AST and TID-14844 source term assumptions. 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. This LAR does not propose to modify the EQ design basis to adopt AST. The PBNP EQ radiation analysis will continue to be based upon TID-14844 assumptions.

5.5 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.

5.6 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 NUREG-0737 Items II.B.2.2 and III.D.3.4, shielding was installed at the C-59 control panel in the PBNP PAB, at two 480 V motor control centers (1B32 and 2B32), and at wall penetrations between the control building and the PAB. Portable shielding was also provided for the CR windows. In support of this license amendment, plant modifications will be performed to replace the CR portable shielding with permanent shielding. Following installation of the permanent shielding modification for the CR that replaces the temporary shielding, the NUREG-0737 Item III.D.3.4 commitment will be eliminated.

Therefore, there is no increase in risk associated with calculation of post-accident access doses via adoption of AST.

5.7 No Significant Hazards Consideration

In accordance with the requirements of 10 CFR 50.90, FPL Energy Point Beach, the licensee, hereby requests amendments to Facility Operating Licenses DPR-24 and DPR-27, for PBNP Unit 1 and Unit 2, respectively.

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.

FPL Energy Point Beach 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 PBNP in accordance with the proposed amendment presents no significant hazards. The FPL Energy Point Beach 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 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 RG 1.183 for the 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 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. The dose consequences in the CR, the exclusion area boundary, and the low population zone do not exceed the regulatory limits provided by the NRC in 10 CFR 50.67 and Regulatory Guide 1.183 for the AST methodology.

   For the locked rotor event, an NRC approved methodology RAVE (Westinghouse WCAP-16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis,") is used to determine rods in DNB. The use of an NRC approved methodology provides an input assumption to the radiological dose consequences calculations. The use of the new methodology does not change the sequence or progression of the accident scenario.

   The proposed TS changes reflect the plant configuration that is required to implement the AST analyses. The equipment affected by the proposed changes is mitigating in nature and relied upon after an accident has been initiated. The operation of various filtration systems, the RHR and the CS systems, including associated support systems, has been considered in the evaluations of these proposed changes. The operation of this equipment has been evaluated for emergency diesel generator loading and fuel consumption. The evaluation demonstrated that the diesel generator loading and fuel consumption do not exceed the diesel generator criteria. 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 DBA.

The operation of containment spray on sump recirculation has been evaluated for increased strainer blockage or reduction in flow from the sump. The evaluation demonstrated that the increase in containment spray will not adversely affect the operation of the emergency core cooling systems during the sump recirculation phase of a DBA.

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. The changes proposed in this LAR involve the use of a new analysis methodology and related regulatory acceptance criteria. The proposed TS changes reflect the plant configuration that is required to implement the AST analyses. No new or different accidents result from utilizing the proposed changes. Although the proposed changes require modifications to the VNCR system, as well as modifications to the RHR system and CS system, these changes will not create a new or different kind of accident since they are related to system capabilities that provide protection from accidents that have already occurred. The operation of this equipment has been evaluated for emergency diesel generator loading and fuel consumption. The evaluation demonstrated that the diesel generator loading and fuel consumption do not exceed the diesel generator criteria.

The operation of containment spray on sump recirculation has been evaluated for increased strainer blockage or reduction in flow from the sump. The evaluation demonstrated that the increase in containment spray will not adversely affect the operation of the emergency core cooling systems during the sump recirculation phase of a DBA.

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 FSAR nor do they introduce any unique precursor mechanisms.

For the LR event, an NRC approved methodology RAVE (Westinghouse WCAP-16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis,") is used to determine rods in DNB. The use of an NRC approved methodology provides an input assumption to the radiological dose consequences calculations. The use of the new methodology does not alter the nature of events postulated in the FSAR 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. The changes proposed in this license amendment involve the use of a new analysis methodology and related regulatory acceptance criteria. The proposed TS changes reflect the plant configuration that is required to implement the AST analyses. 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, no significant reduction of margin has occurred and analyses adequately bound postulated event scenarios. The proposed changes continue to ensure that the dose consequences of DBAs at the exclusion area and LPZ boundaries and in the CR 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.

For the LR event, an NRC approved methodology RAVE (Westinghouse WCAP-16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis,") is used to determine rods in DNB. The use of an NRC approved methodology provides an input assumption to the radiological dose consequences calculations. The use of the new methodology does not reduce any margins of safety for the LR event; therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL Energy Point Beach 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.

5.8 Conclusion

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.

6.0 ENVIRONMENTAL CONSIDERATION

Adoption of the AST and associated TS changes, which justify 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 changes, 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.

FPL Energy Point Beach has determined that operation with the proposed license 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 license 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 license amendment.

7.0 PRECEDENTS

NRC has previously approved implementation of the AST methodology at a number of other nuclear power stations. PBNP is crediting certain aspects of the following previously approved AST submittals as described below. Additional detail is provided in Enclosure 3 of this LAR.

For the LOCA 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 (Adams Accession No. ML003727500) (Reference 8.18) and Shearon Harris (Adams Accession No. ML012830516) (Reference 8.19) for the application of the AST methodology. Additional detail is provided in Enclosure 3 section 6.1.

For the CRDE analysis, the primary-to-secondary leakage and the steaming from the SGs is assumed to continue until primary system pressure is less than secondary system pressure. This is assumed to occur at 2000 seconds. This assumption is consistent with previously approved submittals for Indian Point Unit 2 (Adams Accession No. ML003727500) (Reference 8.18), Shearon Harris (Adams Accession No. ML012830516) (Reference 8.19), and Kewaunee Power Station (Adams Accession No. ML070430020) (Reference 8.20). Additional detail is provided in Enclosure 3 section 6.5.

The current FHA analysis assumes gap fractions of 8% for 1-131, 10% for Kr-85 and 5% for all other noble gas and iodine nuclides. This LAR proposes a change for the assumed FHA gap fractions to 12% for I-131, 30% for Kr-85, and 10% for the other iodines and noble gases. This change reflects the assumption that the damaged nuclear fuel used in the FHA radiological analysis is assumed to exceed the RG 1.183, Table 3, Footnote 11 criteria. The gap fractions are those from Regulatory Guide 1.25 with the value for I-131 increased by 20%, consistent with the recommendation in NUREG/CR-5009. This is a conservative assumption and follows the method approved by the NRC for Kewaunee Power Station (Adams Accession No. ML070430020) (Reference 8.20). Additional detail is provided in Enclosure 3 sections 3.3 and 6.6.

8.0 REFERENCES

8.1 Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000. (ML003716792).

8.2 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.

8.3 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. (ML040680918)

8.4 WCAP-16259-P-A (Proprietary), "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," Beard, C.L., et al., August 2006. (ML062580352)

8.5 WCAP-10965-P-A (Proprietary), "ANC – A Westinghouse Advanced Nodal Computer Code," Y. S. Liu, et al., September 1986.

8.6 WCAP-14565-P-A (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Sung, Y.X., et al., October 1999. (ML993160101)

8.7 WCAP-14882-P-A (Proprietary), "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," Huegel, D.S., et al., April 1999.

8.8 Nuclear Energy Institute (NEI) 99-03, "Control Room Habitability Assessment Guidance," Revision 1, Nuclear Energy Institute, Washington, DC, March 2003.

8.9 Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2003-01, "Control Room Habitability," June 12, 2003 (ML031620248).

8.10 NRC Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," March 7, 2006.

8.11 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. (ML030970703)

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

8.13 NRC Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)," March 1972.

8.14 NRC NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988.

8.15 NRC Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance"

8.16 USNRC Letter, "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 (Tac Nos. MC7650 MC7651)," September 23, 2005. (ML052560089)

8.17 USNRC Letter, "NUREG-0737 Item II.D.3.4 – Control Room Habitability at Point Beach Nuclear Plant Units 1 and 2," August 10, 1982.

8.18 USNRC Letter, "Indian Point Nuclear Generating Unit No. 2 – Re: Issuance of Amendment Affecting Containment Air Filtration, Control Room Air Filtration, and Containment Integrity During Fuel Handling Operations (Tac No. MA6955)," July 27, 2000. (ML003727500)

8.19 USNRC Letter, "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Re: Steam Generator Replacement and Power Uprate (Tac Nos. MB0199 and MB0782)," October 12, 2001. (ML012830516)

8.20 NRC Letter to Kewaunee Power Station, "Issuance of Amendment Re: Radiological Accident Analysis and Associated Technical Specifications Change (TAC No. MC9715)," dated March 8, 2007. (ML070430020)

ENCLOSURE 2

LICENSE AMENDMENT REQUEST 241
TECHNICAL SPECIFICATION AND BASES PAGES
MARK UP

POINT BEACH NUCLEAR PLANT

40 Pages Follow

1.1 Definitions

L_a

The maximum allowable primary containment leakage rate, L_a, shall be ~~0.4~~0.2% 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 steps.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 >50 μCi/gm.	C.1 AND C.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	---NOTE--- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT Xe-133 Specific Activity ≤ 520 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

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 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

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.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$.

2. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $\leq 0.6\ L_a$ for the combined Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests.

3. Air lock testing acceptance criteria are:

i. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

ii. For each door seal, leakage rate is equivalent to $\leq 0.02\ L_a$ at $\geq P_a$ when tested at a differential pressure of $\geq$ to 10 inches of Hg.

e. The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.6 Reporting Requirements

5.6.4 CORE OPERATING LIMITS REPORT (COLR) (continued)

(4) WCAP-14787-P, Rev. 2, "Revised Thermal Design Procedure Instrument Uncertainty Methodology for Wisconsin Electric Power Company Point Beach Units 1 & 2 (Fuel Upgrade & Uprate to 1656 MWt-NSSS Power with Feedwater Venturis, or 1679 MWt-NSSS Power with LEFM on Feedwater Header), October, 2002 (approved by NRC Safety Evaluation, November 29, 2002).

(5) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," August 1985.

(6) WCAP-10054-P-A, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," Addendum 2, Revision 1, July 1997.

(7) WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

(8) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control," Revision 1A, February 1994.

(9) WCAP-10924-P-A, "Large Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," and Addenda, December 1988. (cores not containing 422 V+ fuel)

(10) WCAP-10924-P-A, "LBLOCA Best Estimate Methodology: Model Description and Validation: Model Revisions," Volume 1, Addendum 4, August 1990. (cores not containing 422 V+ fuel)

(11) Caldon, Inc., Engineering Report-80P, "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997.

(12) Caldon, Inc., Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ System," Revision 0, May 2000.

(13) WCAP-16259 P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis."

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

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~~ Accident Source Term~~"~~ (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

SAFETY LIMIT VIOLATIONS (continued)

and create a potential for radioactive releases in excess of 10 CFR ~~100, "Reactor Site Criteria," limits~~50.67 "Accident Source Term" (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, "Accident Source Term".
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

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,"~~ 50.67, "Accident Source Term" 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~~10 CFR 50.67, "Accident Source Term".
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, "Accident Source Term" 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~~s~~ of 10 CFR 50.67, "Accident Source Term" (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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.13

SR 3.3.1.13 is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, SI Input from ESFAS, and the Condenser Pressure-High and Circulating Water Pump Breaker Position inputs to the P-9 Interlock. This TADOT is performed every 18 months. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function for the Reactor Trip Breakers and the undervoltage trip circuits for the Reactor Trip Bypass Breakers.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to exceeding the P-9 interlock whenever the unit has been in MODE 3. This Surveillance is not required if it has been performed within the previous 31 days. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to exceeding the P-9 interlock.

SR 3.3.1.15

SR 3.3.1.15 is the performance of an ACTUATION LOGIC TEST on the RCP Breaker Position (Two Loop), Reactor Coolant Flow-Low (Two Loop) and Underfrequency Bus A01 and A02 Trip Functions, and P-6, P-7, P-8, P-9 and P-10 Interlocks every 18 months.

The 18 month frequency is based on the need to perform this surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance were performed with the reactor at power.

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. 10 CFR 50.67, "Accident Source Term"

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 or a 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~~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 ~~make-up~~ mode (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~~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, "Accident Source Term" (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, "Accident Source Term".

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in ~~10 CFR 100.11~~10 CFR 50.67, "Accident Source Term" (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

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

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT Xe-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 2).

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of 500 gallons per day per steam generator exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of ~~1.0~~0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.13, "Secondary Specific Activity."

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine 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~~, by a factor of 500,~~ the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500) or SGTR (by a factor of 335) respectively. The second case assumes the initial reactor coolant iodine activity at 50.0 μCi/gm DOSE EQUIVALENT I-131 due to an iodine spike caused by a reactor or RCS transient prior to the accident. In both cases, the noble gas specific activity is assumed to be

BASES

LCO

The iodine specific activity in the reactor coolant is limited to ~~0.8~~0.5 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 520 μCi/gm DOSE EQUIVALENT Xe-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2).

The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

APPLICABILITY

In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT Xe-133 is necessary to limit the potential consequences of a SLB or SGTR to within the SRP acceptance criteria (Ref. 2).

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

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is ≤ 50.0 μCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(s), relying on Required Action A.1 and A.2 while DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT Xe-133 is not detected, it should be assumed to be present at the minimum detectable activity.

A Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be preformed in those MODES, prior to entering MODE 1.

SR 3.4.16.2

This surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas 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 iodine spike initiation; samples at other times would provide inaccurate results.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be preformed in those MODES, prior to entering MODE 1.

REFERENCES

1. ~~10 CFR 100.11~~10 CFR 50.67, "Accident Source Term".

2. Standard Review Plan (SRP), Section ~~15.1.5 Appendix A (SLB) and Section 15.6.3 (SGTR)~~15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms," Rev 0, July 2000.

3. FSAR, Section 14.2.4.

4. FSAR, Section 14.2.5.

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) and 10 CFR ~~100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits)~~50.67, "Accident Source Term" (Ref. 3) dose guideline limit.

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, "Accident Source Term".
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 or 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 ≤ 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 and purge supply/exhaust flanges, 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 (~~Reference 5~~Ref. 6). 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 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. 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 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).

BASES

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, "Accident Source Term" (Ref. 3) limits.

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*)

This test is conducted in MODE 2 under low steam flow conditions (≤ 5% steam flow) at operating temperature and pressure. 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 a delay of testing to 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 ~~100.11~~50.67, "Accident Source Term".
4. ASME Boiler and Pressure Vessel Code, Section XI, OM Code, Code for Operation and Maintenance of Nuclear Power Plants.
5. TRM 4.7, Inservice Testing Program.
6. Standard Review Plan (SRP) 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms," Rev 0, 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, however only Mode 5 is credited in the FSAR Chapter 14 radiological analyses. FSAR 9.8 (Ref. 1) provides a system description for all five modes of CREFS 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.~~

~~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~~

BASES

BACKGROUND (continued)

~~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) - Operation in this mode is similar to mode 3 except return air inlet damper VNCR-4851B to the emergency 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 prevent 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 manually initiated from the control room. 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.~~

~~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.~~
Mode 5 (emergency HEPA/charcoal filtered outside air and HEPA/charcoal filtered return air) allows a combination of outside air

BASES

and return air >1955 cfm 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 CREF 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 the Technical Specification LCO.

BASES

BACKGROUND (continued)

The limiting design basis accident for the control room dose analysis is the large break LOCA. CREFS ~~does not~~ automatically ~~restart~~restarts 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 (~~m~~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 total effective dose equivalent (TEDE) limit of 5~~thyroid dose limit of 30~~ 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~~Both recirculation fans (W-13B1 ~~or~~ and W-13B2) ~~is~~are 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 automatically and manually initiated in the emergency ~~make-up~~ mode of operation (~~m~~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

SR 3.7.9.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each fan subsystem once every month provides an adequate check of this system. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the reliability of the equipment.

SR 3.7.9.2

This SR verifies that the required CREFS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The Frequency of CREFS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.9.3

This SR verifies that each CREFS emergency ~~make-up~~ fan starts and operates on an actual or simulated actuation signal. The Frequency of 18 months is specified in Regulatory Guide 1.52 (Ref. 3).

SR 3.7.9.4

This SR verifies that each CREFS automatic damper in the emergency ~~make-up~~ mode flow path will actuate to its required position on an actuation signal. The Frequency of 18 months is specified in Regulatory Guide 1.52 (Ref. 3).

SR 3.7.9.5

This test verifies manual actuation capability for CREFS. Manual actuation capability is a required for OPERABILITY of the CREFS. The 18 month Frequency is acceptable based on the inherent reliability of manual actuation circuits.

SR 3.7.9.6

This SR verifies the integrity of the control room enclosure. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CREFS. During the emergency mode of operation, the CREFS is

BASES

SURVEILLANCE REQUIREMENTS (continued)	designed to pressurize the control room ≥ 0.125 inches water gauge 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, Accident Source Term (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~~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

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, Accident Source Term (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~~60 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~~0.5 μCi/gm. The duration of the accident-initiated iodine spike is assumed to be ~~1.6~~4 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.~~ at the Technical Specification limit of 520 μCi/gm DE Xe-133

3. The secondary coolant iodine activity is assumed to be ~~1.0~~ 0.1 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~~ 3.5x 10^{-4} m^3/sec.

7. The SG connected to the ruptured main steam line is assumed to boil dry within ~~30~~2 minutes.

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.

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

11. 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.

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

13. 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.~~

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

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

LCO (continued)

specified in 10 CFR 50.67, Accident Source Term (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.

BASES

BASES

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.

REFERENCES

1. ~~10 CFR 100.11~~.10 CFR 50.67, "Accident Source Term"

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~~15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July ~~1981~~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. These Surveillances performed during MODE 6 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 15.7.4, Rev. 1, July 1981.~~
2 Standard Review Plant (SRP) Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms," Rev 0, July 2000.

3. 10 CFR 50.67, "Accident Source Term".

4. Regulatory Guide 1.183 (Rev. 0).

5.NRC SE dated 02/19/2008 for LAR 249

ENCLOSURE 3

LICENSE AMENDMENT REQUEST 241
TECHNICAL EVALUATION

POINT BEACH NUCLEAR PLANT (PBNP)

89 Pages Follow

References

Tables

Figures

Appendix A Input Assumptions Comparison Tables

89 Pages Follow

1.0 Introduction

1.1 Evaluation Overview and Objective

The purpose of this technical evaluation is to provide the results of the analyses that determined that the Point Beach Nuclear Plant (PBNP) can safely operate with alternative source term (AST). This evaluation documents the PBNP full implementation of the AST in accordance with 10 CFR 50.67 (Reference 8.1), as described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 8.2). The offsite and control room (CR) dose analyses for the following PBNP Final Safety Analysis Report (FSAR) accidents have been reanalyzed. These analyses also satisfy the CR habitability requirements of Generic Letter (GL) 2003-01 (Reference 8.3).

The following FSAR Chapter 14 accidents are analyzed:

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

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

The proposed license amendment revises the PBNP Technical Specifications (TS) that are affected by the proposed revision to the Design Basis Accident (DBA) source term. This amendment will remove the need for post-accident reliance on potassium iodide (KI) for CR operators from the PBNP licensing basis.

The FHA, based on the AST, was submitted previously under Reference 8.4 and approved under Reference 8.5. The revised FHA dose analysis presented in this license amendment request (LAR) determined that the exclusion area boundary (EAB), low population zone (LPZ), and the CR doses meet RG 1.183 acceptance criteria. Input assumption changes are discussed in Section 6.6 of this enclosure.

1.2 Summary and Conclusions

The offsite and CR doses, for the accidents described in Chapter 14 of the PBNP FSAR, have been reanalyzed consistent with the AST methodology described in RG 1.183. The calculated doses for the EAB, LPZ and CR are all within the acceptance criteria of 10 CFR 50.67, as shown in the table below:

	Limiting two hours	Duration of Activity Releases		30-day Dose	
Accident	**EAB (rem TEDE)**	**LPZ (rem TEDE)**	**Offsite Dose Criteria (rem TEDE)**	**CR (rem TEDE)**	**CR Dose Criteria (rem TEDE)**
LOCA	14.2	1.6	25	4.9	5.0
SGTR - Pre-Accident Spike	2.0	0.2	25	1.9	5.0
SGTR - Accident Initiated Spike	0.6	0.1	2.5	0.5	5.0
LR	2.0	0.5	2.5	4.6	5.0
MSLB - Pre-Accident Spike	0.14	0.03	25	1.9	5.0
MSLB - Accident Initiated Spike	0.20	0.08	2.5	4.0	5.0
CRDE	2.3	0.8	6.3	2.9	5.0
FHA	2.7	0.2	6.3	4.3	5.0
RVHD	0.1	0.1	6.3	0.5	5.0

Assumed CR unfiltered inleakage of 200 cfm for LOCA and MSLB, and 300 cfm for the remaining Non-LOCA events provides acceptable CR doses. Measured CR inleakage meets these criteria.

Based on the information presented in this LAR, FPL Energy Point Beach has 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.

Based on these results, the need for post-accident reliance on KI for CR operators will be removed from the PBNP licensing basis.

1.3 Changes to the PBNP Design and Licensing Basis

The following denotes the proposed changes to the PBNP design and licensing basis required for full-implementation of this AST:

- 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 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 CR intake are re-analyzed for existing pathways using NUREG/CR-6331, "ARCON96; Atmospheric Relative Concentrations in Building Wakes," May 1997 (Reference 8.7);
- New values for CR unfiltered air inleakage assumed to bound the test results are modeled;
- CR ventilation system will operate with filtered recirculation in addition to filtered outside air intake (including auto-loading of the control room emergency filtration system (CREFS) fans onto their associated emergency diesel generator during a loss of offsite power (LOOP) coincident with a LOCA);
- Credit is taken for future shielding modifications to the CR;
- Reduced values in the allowable dose equivalent (DE) I-131 concentrations in the primary and secondary systems are used;
- A reduced allowable containment leakage is modeled;
- A factor of two increase over FPL Energy Point Beach's commitment to Item III.D.1.1 of NUREG-0737 is applied to the emergency core cooling system (ECCS) leakage limit for CR habitability radiological analyses;
    - ECCS recirculation leakage into the primary auxiliary building (PAB) is assumed to be 300 cc/min
    - ECCS back-leakage to the refueling water storage tank (RWST) is assumed to be 500 cc/min
- Credit is taken for containment spray (CS) removal of fission products while on ECCS recirculation (LOCA);
- Flashing fractions are applied to the SGTR break flow;
- The need for post-accident reliance on KI for CR operators is removed from the PBNP licensing basis;
- The requirement that the reactor must be shut down for 100 hours prior to lifting the reactor vessel head is eliminated based on the revised RVHD analysis;
- Revised radial core power peaking factors are applied to the radiological analyses;
- FHA gap fractions have been increased to reflect the fact that some nuclear fuel assemblies exceed the Table 3, Footnote 11 criteria of RG 1.183. As a conservative approach, the gap fractions are those from RG 1.25 (Reference 8.8) with the value for I-131 increased by 20%, consistent with the recommendation in NUREG/CR-5009 (Reference 8.9);

- For the CRDE, the primary to secondary leak rate duration was determined based on time required for the primary system pressure to decrease to the secondary system pressure. The primary system pressure decrease was based on a small-break LOCA (about 2" diameter). The depressurization time has been conservatively increased by 25% above the calculated depressurization time;

- For the LR event, 30% of the fuel rods in the core are assumed to suffer damage due to DNB. The RAVE methodology "Westinghouse WCAP-16259-P-A" is being incorporated into the CLB for the LR event to determine rods in DNB. RAVE is a methodology that uses inputs from three other Westinghouse codes that have been approved for use by the NRC: RETRAN, ANC (SPNOVA) and VIPRE;

- Credit is taken for manual operator action to restore PAB ventilation (VNPAB) within 30 minutes following the alignment of residual heat removal (RHR) to containment sump recirculation mode of operation. If a LOCA occurs coincident with a LOOP, the VNPAB will be manually restarted to ensure that the auxiliary building vent stack is the source of the release associated with the ECCS leakage phase of the event. This configuration is an input assumption for the LOCA radiological analysis.

1.4 RG 1.183 Implementation Provisions and Deviations

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

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

- This LAR proposes to continue crediting the current control room ventilation (VNCR) system. The VNCR system is classified as non-safety related, however selected components such as the control room charcoal filter fans, the control room recirculation fans, damper controllers and the charcoal/HEPA/roughing filter have been upgraded to augmented quality status. For the accident mode of operation, the two radiation monitor actuation signals are diverse and the radiation monitors are augmented quality status. The containment isolation actuation signals are safety-related and redundant. The fans, dampers and filter are tested in accordance with Surveillance Requirements SR 3.7.9.1 through SR 3.7.9.6 of the Point Beach Technical Specifications (see page 3.7.9-2 of Enclosure 2). In order to address the requirements of Section 5.1.2 of Regulatory Guide 1.183, the system will be modified to ensure redundancy of active components and to auto-start the system from the safety related diesel generators on loss of offsite power. The current radiological dose analyses presented in the PBNP FSAR credit the operation of this system in calculating the dose to the CR operators.

- 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 recommends an 8-hour spike, but allows for shorter durations if it can be shown that the activity released by an 8-hour spike exceeds the available gap activity. As such, the MSLB assumes a spike of 4 hours, rather than 8 hours.

- This LAR proposes a deviation from RG 1.183, Section 1.1.2 with respect to manual operator action. New CR operator actions to align core injection and CS flow to preset throttled positions are required to realign CS from injection to recirculation. The dose projections for the LOCA radiological analysis assume that CS is maintained throughout the injection phase, and continued for 3-hours during the ECCS recirculation phase. There will be no more than a 20-minute spray interruption to switch from injection to recirculation spray. The ability to establish and maintain CS during the early recirculation phase is essential, as this is modeled as the period of highest activity release from a postulated damaged core.

  Once the RHR system is aligned to take pump suction from the containment sump, the operator actions to align the RHR and CS systems for recirculation spray will be accomplished from the CR, with no local operator action required. The controls and instrumentation necessary to place the CS system in recirculation spray operation will be provided in the CR. Throttling of the CS flow path (flow limiting orifice) and throttling of the RHR core injection flow path (preset throttle position of SI -852A/B) will minimize the potential for operator error, while ensuring adequate net positive suction head (NPSH) is maintained to the running RHR pump, ensuring adequate core injection flow is available at all times, and allowing sufficient time to place recirculation spray in service to meet radiological analyses assumptions associated with implementation of AST. RHR injection flow is maintained greater that 500 gpm during the alignment and following initiation of containment spray on recirculation. CS flow on recirculation is greater than 900 gpm. The recirculation spray flow path is illustrated in Figure 3.

  The spray additive tank discharge valves are air-operated valves, remotely operated from the CR. If instrument air is unavailable, the spray additive tank discharge valves can be manually closed by an operator in the PAB. The manual closure can be performed within the required time and the dose to the operator has been determined to be acceptable.

  The alignment to recirculation spray to ensure that the necessary actions can be accomplished well within 20 minutes has been demonstrated on the PBNP simulator by operations and training department personnel. Procedures will be revised and training for all operating shifts will be completed prior to implementation of the AST license amendment.

  Alignment to CS on recirculation has no impact on the existing manual operator actions to place RHR on sump recirculation, as the line-up to provide a suction source from RHR to the CS for recirculation spray operation occurs after one or both trains of RHR suction have been transferred to the containment sump for the recirculation phase. The CR operator actions, to transfer RHR suction from the RWST to the containment sump, and the local operator actions to align cooling to the RHR heat exchangers for the recirculation phase, will remain unchanged.

- This LAR proposes an additional deviation from RG 1.183, Section 1.1.2 with respect to manual operator action of the VNPAB system, which is classified as non-safety related. VNPAB fans are powered by safety-related power supplies with diesel generator backup. New CR operator actions are required to restore VNPAB within 30 minutes following the alignment of RHR to containment sump recirculation mode of operation.

If a LOCA occurs coincident with a LOOP, the VNPAB system will be manually restarted to ensure that the auxiliary building vent stack is the source of the release associated with the ECCS leakage phase of the event. This configuration is an input assumption for the LOCA radiological analysis.

Once the RHR system is aligned to take pump suction from the containment sump, the operator actions to align the VNPAB system will be accomplished from the CR, with no local operator action required. The controls necessary to place the VNPAB ventilation system in service will be provided in the CR.

Procedures will be revised and training for all operating shifts will be completed prior to implementation of the AST license amendment.

Manual restoration of the VNPAB system following the alignment of RHR on sump recirculation has no impact on the existing manual operator actions to place RHR on sump recirculation.

- For the CRDE, the primary-to-secondary leak rate duration was determined based on time required for the primary system pressure to decrease to the secondary system pressure. The primary system pressure depressurization was based on a 2" diameter LOCA. The depressurization time in the CRDE dose analysis has been conservatively increased by 25% above the calculated depressurization time.

- FPL Energy Point Beach is proposing to implement alternative gap fractions for the FHA. This change reflects the fact that fuel rods are assumed to exceed the Table 3, Footnote 11 criteria of RG 1.183. As a conservative approach, the assumed gap fractions are those from RG 1.25 with the value for I-131 increased by 20%, consistent with the recommendation in NUREG/CR-5009 (Reference 8.9). This is a conservative assumption and follows the method approved by the NRC for Kewaunee Power Station (Adams Accession No. ML070430020) (Reference 8.6).

- 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. 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. This LAR does not propose to modify the EQ design basis to adopt AST. The PBNP EQ radiation analysis will continue to be based upon TID-14844 assumptions.

1.5 Computer Codes

	Code	Application
1	Industry computer code, "ORIGEN-S: Scale System Module to Calculate Fuel Depletion, Actinide, Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms". (ORNL/TM-2005/39)	ORIGEN-S is an isotope production and depletion code. This is the industry standard code that uses regularly updated data.
2	NRC sponsored code, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03 (NUREG/CR-6604, Supplement 2)	Used to calculate activity transfers, decay, depletion, releases, and resulting doses.
3	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 CR operator dose by modeling source-shield-detector configurations.
4	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 CR, and the CR charcoal and high HEPA filters to determine CR operator dose.
5	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.
6	NRC Sponsored Code ARCON96, "Atmospheric Relative Concentrations in Building Wakes," developed by Pacific Northwest Laboratory (NUREG/CR-6331, Revision 1)	Used to calculate atmospheric dispersion factors (χ/Q) for CR doses.
7.	Westinghouse Proprietary Computer Code, FIPCO Version 3.1	Used to calculate reactor coolant activity based on 1% fuel defect level.
8.	Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis - RETRAN, ANC (SPNOVA) and VIPRE (RAVE)	Used to determine rods-in-DNB for the LR event[(1)].

Notes:

(1) WCAP-10965-P-A ANC (SPNOVA) (Reference 8.10), WCAP-14565-P-A VIPRE (Reference 8.11), WCAP-14882-P-A RETRAN (Reference 8.12), and RAVE (Reference 8.13) were generically approved for use at Westinghouse plants and are applicable to PBNP. SPNOVA is part of the ANC code which is included in the PBNP CLB. Since PBNP is the first plant to use this methodology, Enclosure 5 (proprietary) and Enclosure 6 (non-proprietary) of this submittal address the conditions and limitations addressed in the safety evaluation report (SER) for WCAP-16259-P-A.

Enclosure 5 contains the affidavit from Westinghouse Electric Company for withholding proprietary information. Enclosure 7 provides the SER Compliance Matrices for ANC (SPNOVA), (WCAP-10965-P-A), VIPRE, (WCAP-14565-P-A), and RETRAN, (WCAP-14882-P-A) and addresses the conditions and limitations associated with the SER for WCAP-10965-P-A, WCAP-14565-P-A and WCAP-14882-P-A.

2.0 Radiological Evaluation

2.1 Introduction

The PBNP licensing basis for the radiological consequences analyses currently utilizes methodologies, assumptions and dose limits that are derived from TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," (Reference 8.14). RG 1.183 (Reference 8.2) provides guidance for application of AST accident source terms used in DBA radiological consequences analyses, as allowed by 10 CFR 50.67 (Reference 8.1). The AST methodology, as established in RG 1.183, is being used to calculate the offsite and CR radiological consequences for PBNP to support the increase in the assumed CR unfiltered inleakage, and to eliminate the reliance on KI for the CR operators from the PBNP licensing basis. The following accidents are analyzed: LOCA, SGTR, LR, MSLB, CRDE, FHA, and RVHD. Each accident and the specific input assumptions are described in detail in subsequent sections in this enclosure.

The FHA has been previously analyzed using the AST methodology. NRC approval was given to PBNP via SER dated April 2, 2004 (Reference 8.5). The revised dose analysis for the FHA presented in this LAR determined that the EAB, LPZ and the CR doses meet RG 1.183 acceptance criteria. Input assumption changes are discussed in Section 6.6 of this enclosure.

Offsite and CR 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 this enclosure.

The current licensed maximum reactor core power level is 1540 MWt. The analyses in this enclosure model a maximum core power of 1800 MWt with 0.6% uncertainties (analyzed core power of 1811 MWt). This power level was chosen to support a future extended power uprate (EPU) license amendment request and is conservative with respect to the current licensed power level. Although the analyses were performed at a higher power level, this LAR is not requesting approval for use of the higher core power level.

2.2 Common Analysis Inputs and Assumptions

Common analysis input assumptions include those for the VNCR system and dose calculation model, radiation source terms, and atmospheric dispersion factors. The accident specific inputs and assumptions are discussed in Sections 6.1-6.7 of this enclosure.

The TEDE doses are determined at the EAB for the worst 2-hour interval. The TEDE doses at the LPZ are determined for the duration of the event. The TEDE doses for the CR 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 CR, which continue for as long as the activity is circulating within the CR envelope (CRE).

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 Environmental Protection Agency (EPA) Federal Guidance Report No. 11 (Reference 8.15) and are given in Table 1. The DCFs used in determining the EDE dose are from EPA Federal Guidance Report No. 12 (Reference 8.16) 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 doses are those in the PBNP current licensing basis (CLB).

Parameters used in the CR personnel dose calculations are provided in Table 4. These parameters include the normal operation flow rates, the post-accident operation flow rates, CR volume, filter efficiencies, and CR operator breathing rates. Atmospheric dispersion factors are specific to each analysis and are calculated with respect to the location of the CR 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 provided in Tables 18 through 24.

Unfiltered inleakage into the CR is assumed to be 200 cfm for LOCA and MSLB, and 300 cfm for the remaining non-LOCA events. CLB analyses for CR habitability assume an unfiltered inleakage of 10 cfm based on the guidance of the Murphy-Campe methodology (Reference 8.17). Recent industry inleakage testing of the CRE has shown that 10 cfm may not be a conservative value for this parameter (Reference 8.3). This early methodology provided a value of 10 cfm unfiltered inleakage for a pressurized CR.

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 to reduce overall operator dose post-accident. Tracer gas testing for the CR in the proposed VNCR accident mode alignment performed in September 2003 identified 77 ±94 cfm inleakage (Reference 8.25). The CRE and habitability are 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 CR, 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 CR. The half life for each nuclide is provided in Table 7.

The accident-induced primary-to-secondary leak rate, modeled in the LR, MSLB, SGTR and CRDE dose analyses, is significantly greater than the operational leak rate. The assumed accident-induced leak rate is 0.7 gpm, total (TS 5.5.8, "Steam Generator (SG) Program," Accident induced leakage performance criterion of 500 gpd per SG). The assumed density, for conversion to a mass leak rate, is 47 lbm/ft^3 (full power operation), and the corresponding leak rate is 1000 gm/min per SG.

The operational leak rate is 150 gpd per SG and is measured at room temperature (TS Bases 3/4 4.6.2, "Operational Leakage"). The assumed density, for conversion to a mass leak rate, is 62.4 lbm/ft^3, and the corresponding leak rate is 394 gm/min per SG.

The core activity is provided in Table 5. The reactor and secondary coolant concentrations are provided in Table 6. The core and coolant activities in Tables 5 and 6 are based on an analyzed core power of 1811 MWt. The core and 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:

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

Where:

D_{CEDE} = CEDE dose via inhalation (rem).
DCF_i = CEDE DCF via inhalation for isotope i (Sv/Bq) (Table 1)
$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j (Bq)
$(BR)_j$ = 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)
F = units conversion factor 0.01

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

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

Where:

D_{EDE} = external exposure dose via cloud submersion (rem)
DCF_i = EDE DCF via external exposure for isotope i (Sv·m^3/Bq·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)
F = units conversion factor 0.01

Control Room Dose Calculation Models

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

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

CR inhalation doses are calculated using the following equation:

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

Where:

D_{CEDE} = CEDE dose via inhalation (rem)
DCF_i = CEDE DCF via inhalation for isotope i (Sv/Bq) (Table 1)
$Conc_{ij}$ = concentration in the CR of isotope i, during time interval j, calculated dependent upon inleakage, filtered inflow, total outflow and CR volume ($Ci\text{-}sec/m^3$)
$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table 4)
$(OF)_j$ = occupancy factor during time interval j (Table 4)
F = units conversion factor 0.01

CR external exposure doses due to activity in the CR volume are calculated using the following equation:

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

Where:

D_{EDE} = external exposure dose via cloud submersion in rem.
GF = geometry factor, calculated based on Reference 8.17, using the equation:

$GF = \frac{1173}{V^{0.338}}$, where V is the CR volume in ft^3

DCF_i = EDE DCF via external exposure for isotope i ($Sv \cdot m^3/Bq \cdot sec$) (Table 2)
$Conc_{ij}$ = concentration in the CR of isotope i, during time interval j, calculated dependent upon inleakage, filtered inflow, total outflow and CR volume ($Ci\text{-}sec/m^3$)
$(OF)_j$ = occupancy factor during time interval j (Table 4)
F = units conversion factor 0.01

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 1800 MWt with 0.6% uncertainties (analyzed core power of 1811 MWt), and uprated values of fuel enrichment and burnup. 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.

Core and plant parameters, for four fuel management cycles representing the core inventory during the transition from the current power level to the analyzed power level, are shown in the table below. The results are very similar cycle to cycle, although Cycle C has a slightly higher I-131 inventory which results in the highest LOCA dose, and is bounding for both units during the transition to the uprated power level.

Parameter	Cycle A	Cycle B	Cycle C	Cycle D
Nominal core power level (MWt)	1800	1800	1800	1800
Analyzed core power (0.6% uncertainty) (MWt)	1811	1811	1811	1811
System pressure (psia)	2250	2250	2250	2250
Core average moderator temperature, HZP (°F)	547	547	547	547
Core average moderator temperature, HFP (°F)	581	581	581	581
Core average moderator inlet temperature, HFP (°F)	542.9	542.9	542.9	542.9
Core uranium mass (MTU)	47.606	47.466	47.404	47.404
Uranium mass per assembly (MTU)	0.3934	0.3923	0.3918	0.3918
Cycle burnup (end of full power capability) – EOFPC (MWD/MTU) – EOFPC is the last HFP burnup step	18916	18962	18985	21644
Total cycle burnup (not including coastdown) (EFPD)	500	500	500	570
Boron concentration at EOFPC (ppm)	8	6	12	9
Coastdown length (MWD/MTU)	1135	1138	1140	1139
Total cycle burnup (including coastdown (MWD/MTU)	20051	20100	20125	22783
Total cycle burnup (including coastdown (EFPD)	530	530	530	600

The ORIGEN-S code was used to model the EPU fuel management for the typical cycles, identified as Cycles A through D, for Point Beach Unit 2 using the parameters of loading, burnup, and enrichment for each fuel region. ORIGEN-S edits were selected to provide core inventory by nuclide and element. Cycle C was identified as the most conservative. The reported nuclides are those commonly used in RG 1.183 AST analyses.

The core nuclide inventory includes a multiplier of 1.04, adding margin to the core inventory, to account for cycle-to-cycle variations in enrichment, cycle burnup, and loading that may not have been explicitly addressed in the parametric ranges included in the ORIGEN-S modeling.

3.2 Coolant Inventory

For the reactor coolant system (RCS), maximum coolant activities obtained during a cycle of operation are calculated. Small fuel cladding defects are assumed present, in the initial core loading, and uniformly distributed throughout the core. 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 decontamination factor (DF), and volume control tank noble gas stripping behavior.

Dose equivalent (DE) limits were defined for the concentration of iodine in the RCS in equilibrium and spike conditions, for noble gases in the RCS and for iodine in the SG secondary. The dose equivalence concentrations were determined using the DCFs provided in References 8.15 and 8.16 consistent with the TS definitions for DE I-131 and DE Xe-133.

The following coolant inventories were derived from the 1% defect concentrations provided in Table 6:

- RCS equilibrium iodine concentration limit – 0.5 µCi/g DE I-131.
- RCS iodine concentration limit for pre-accident iodine spike – 60 µCi/g DE I-131.
- RCS equilibrium noble gas concentration limit - 520 µCi/g DE Xe-133.
- RCS alkali metal concentrations – corresponding to 0.5 µCi/g DE I-131.
- Secondary coolant equilibrium iodine concentration limit - 0.1 µCi/g DE I-131.
- Secondary coolant alkali metal concentrations – corresponding to 0.1 µCi/g DE I-131.

3.3 Gap Inventory for Non-LOCA Accidents

The gap fractions listed in Tables 2 and 3 of RG 1.183 serve as a basis for determining available activity in the LOCA, CRDE, RVHD, FHA, and LR radiological analyses. RG 1.183, Table 3, Footnote 11 states that the non-LOCA gap fractions are acceptable, if the peak rod burnup is 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. With the exception of the FHA damaged fuel, no other failed fuel is assumed to exceed the criteria of RG 1.183, Table 3, Footnote 11.

All accidents, with the exception of the FHA, use the gap fractions from RG 1.183. The FHA assumes one fuel assembly is damaged. This change reflects the assumption that fuel in this assembly is assumed to exceed the RG 1.183, Table 3, Footnote 11 criteria. As a conservative approach, the gap fractions are those from RG 1.25 with the value for I-131 increased by 20%, consistent with the recommendation in NUREG/CR-5009. This is a conservative assumption and follows the method approved by the NRC for Kewaunee Power Station (ML070430020) (Reference 8.6).

Accident-specific gap fractions are provided in the table that follows.

Non-LOCA Gap Activity Analysis Assumptions

Reference	RG 1.183, Table 2	RG 1.183, Table-3 Non-LOCA	RG 1.183, Appendix H	RG 1.25	NUREG/CR-5009
Applicable Accidents	RVHD	LR	CRDE		FHA
I-131	0.05	0.08	0.10	0.10	0.12
Other iodine	0.05	0.05	0.10	0.10	0.10
Kr-85	0.05	0.10	0.10	0.30	0.30
Other noble gas	0.05	0.05	0.10	0.10	0.10
Alkali metal	0.05	0.12	0.12	n/a	n/a

3.4 Iodine Spike

Consistent with RG 1.183, two types of iodine spikes are considered for both the SGTR and MSLB accidents.

Pre-accident Spike - A reactor transient has occurred prior to the postulated accident and has raised the primary coolant iodine concentration from the proposed TS 3.4.16 limit equilibrium value of 0.5 μCi/gm of dose equivalent (DE) I-131 to a conservative value of 60 μCi/gm DE I-131. See Table 6.

Accident-Initiated Spike - The primary coolant iodine concentration is initially at the proposed TS 3.4.16 limit, specified as 0.5 μCi/gm DE I-131. Following the primary system depressurization and reactor trip associated with an SGTR or MSLB, an iodine spike is initiated in the primary system. This spike is assumed to increase the iodine appearance rate from the fuel to the coolant to a value of either 335 (SGTR) or 500 (MSLB) times the release rate corresponding to the initial primary system iodine concentration.

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 spike duration was calculated based on the spike appearance rate and activity available in the gaps of the fuel with assumed defects.

The following key assumptions were used to maximize the accident-initiated iodine spike appearance rates:

- Letdown flow maximum (including +10% uncertainty) – 99 gpm
- RCS leakage maximum – 11 gpm
- RCS maximum mass – 1.34E8 gm

The initial RCS iodine activities used in the analysis are presented in Table 6. These values serve as the basis for the iodine appearance rate calculations. The iodine appearance rates used in the analysis are presented below. For the SGTR and MSLB accident initiated spike scenarios, the iodine spike is assumed to persist for 8 hours, and 4 hours, respectively, from the start of the event. For the MSLB, all activity in the gap of the fuel with cladding defects is released within 4 hours. The spike appearance rates are provided below.

APPEARANCE RATES (Ci/min)						
Nuclide	**I-130**	**I-131**	**I-132**	**I-133**	**I-134**	**I-135**
335 times the equilibrium rate (SGTR – 8 hours duration)	0.53	53.9	154.8	107.9	73.0	82.1
500 times the equilibrium rate (MSLB – 4 hours duration)	0.79	80.5	231.0	161.0	109.0	122.5

4.0 Accident Atmospheric Dispersion Factors (χ/Q)

4.1 Meteorological Monitoring Program

The PBNP Meteorological Monitoring System consists of three towers. Two towers are located near the shore of Lake Michigan 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.

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.

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 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 meteorological 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 CR. 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 Meteorological Data Preparation

The meteorological data for September 11, 2000, through September 30, 2005 (inclusive), for the meteorological monitoring system's primary parameters, were reviewed and analyzed to assess their validity and adequacy for estimating dose using dispersion modeling techniques. The data were generally of high quality.

During the period of record, calibrations were performed periodically. If an instrument is found to be out of tolerance, it is adjusted or replaced. In the meantime, data from backup instruments are available, if needed. The meteorological monitoring program relies on periodic site checks of each tower and its instruments in order to confirm proper operation and identify the need for preventive maintenance. Site inspections were found to be effective in identifying problems and anticipating failures. Primary meteorological parameters were evaluated at each five minute recorded interval. Values outside of the reasonable ranges were annotated for future reference and processing. With the exception of differential temperature, the lower bound of the acceptable range was set to the low end of the instrument's range. The upper bound for wind direction limit was set to the instrument's maximum value.

4.3 ARCON96 Input Files

The ARCON96 meteorological data are being provided on CD in a separate submittal to facilitate NRC review of the models used in the analysis. Each parameter for each hour was evaluated to determine if at least 30 minutes of valid data were present. The 30-minute criterion is somewhat more conservative than the 15-minutes specified in RG 1.23. (Reference 8.18) This criterion is judged to better represent the true performance of a meteorological monitoring program. The 30-minute criterion corresponds to U.S. EPA guidance EPA-454/R-99-005, "Meteorological Monitoring Guidance for Regulatory Modeling Applications" (Reference 8.19).

4.4 Offsite Atmospheric Dispersion Factors

The atmospheric dispersion (χ/Q) values for the PBNP EAB and the LPZ are those from the CLB. These values were developed from the guidance provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," (Reference 8.20) and meteorological data collected at the site's primary tower from January 1,1991, through December 31, 1993. FPL Energy Point Beach has assessed these χ/Q values against χ/Q values generated using the PAVAN computer code with the meteorological data collected at PBNP from September 2000 to September 2005 and determined the CLB χ/Q's are conservative. The offsite values are presented in Table 3 and represent the maximum sector χ/Q values.

4.5 Control Room Atmospheric Dispersion Factors

The CR intake χ/Q values are calculated using "ARCON96: Atmospheric Relative Concentrations in Building Wakes" methodology (Reference 8.7). Input data consists of hourly on-site meteorological data, release characteristics, 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 CR air intake and intake height.

The χ/Q values are based on meteorological 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 8.21). 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 PAB.

Historical data for containment penetrations local leak rate tests during refueling outages demonstrated that containment leakage is widely distributed over several penetrations. The overall containment leakage limit (L_a) is significantly greater than any one leak identified during local leak rate testing.

Containment leakage (LOCA and CRDE) is evaluated as a diffuse vertical area source (i.e., the containment can potentially leak anywhere on the exposed surface). In these accidents, 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 diffuse vertical area 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 8.21). 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 CR intake. Both the Unit 1 and Unit 2 χ/Q values were calculated in this manner, however, χ/Q values associated with Unit 2 yield a more conservative atmospheric dispersion factor. All 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 CR fresh air intake. This receptor location is used for unfiltered inleakage. Figure 1 shows Unit 1, Unit 2, and common release locations and Figure 2 shows the PBNP site plan.

- Unit 2 containment wall
- Auxiliary building vent stack (ABVS)
- Unit 2 main steam safety valves (A and B)
- Unit 2 containment façade
- Unit 2 purge stack
- Unit 1 RWST
- Unit 2 RWST

The following assumptions are made for these calculations:

- The plume centerline from each release is conservatively transported directly over the CR air intake;

- All releases are assumed to be under the influence of the containment building wake, 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;

- The main steam safety valves (MSSV)/atmospheric dump valves (ADV) releases are from the approximate center of the discharge vents. No adjustment is made for effective release height;

- 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 8.21;

- The ARCON96 values for surface roughness length (i.e., 0.20 meter) and sector averaging constant (i.e., 4.3) are based on Reference 8.21;

- 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 8.21) with respect to the PBNP configuration.

The ARCON96 input values for each release point are provided in Tables 8 through 15. The χ/Q values for all release locations are provided in the accident specific input tables.

5.0 Control Room Habitability

5.1 Control Room Habitability and Control Room Ventilation (VNCR) System

As noted in PBNP FSAR, Section 1.3, the General Design Criteria (GDC) used during the licensing of PBNP predate those provided today in 10 CFR 50, Appendix A. The origin of the PBNP GDC relative to the Atomic Energy Commission proposed GDC is discussed in the PBNP FSAR, Section 1.3. The PBNP CR 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 CR 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 shut down and maintain safe control of the facility without excessive radiation exposures of personnel.

Although this design criterion is applicable in other areas, such as fire protection, high energy line break (HELB), security, etc., the focus of information provided in this section is solely on radiological habitability of the CR.

The control room envelope (CRE) is located in the control building within the turbine building approximately half-way between Unit 1 and Unit 2. The CRE consists of a shared Unit 1 and Unit 2 CR, the computer room, the reactor engineering room and the associated VNCR system ductwork as it transitions through the mechanical equipment room. The cable spreading room on El. 26' (directly below the CR) and mechanical equipment room El. 60' (directly above the CR) are adjacent to the CRE, however, are not included within the CRE.

The VNCR system is designed to provide heating, ventilation, air conditioning, and radiological habitability for the control and computer rooms, both of which are within the CRE. For radiological habitability, the system is capable of providing CR 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-Three Mile Island (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 (Reference 8.22), Item III.D.3.4, while taking credit for use of KI to reduce the thyroid dose.

Implementation of AST eliminates the use of KI to reduce the thyroid dose for personnel in the CRE from the PBNP licensing basis. 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% with a minimum recirculation flow of 1955 cfm; 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.

As a part of PBNP's response to NUREG-0737, Item III.D.3.4, "Control Room Habitability Requirements," PBNP committed to the placement of portable lead shielding to limit radiation exposure in the CRE (Reference 8.31). Current radiological analyses assume placement of 0.5 inches of lead-equivalent thickness shielding for the window and 0.25 inches of lead-equivalent thickness shielding for the doors to reduce the post-accident dose from the passing plume. Permanent shielding is being installed for the CRE that replaces the temporary shielding installed in response to NUREG-0737 Item III.D.3.4. Once installed, PBNP's regulatory commitment to NUREG-0737, Item III.D.3.4 "Control Room Habitability Requirements" to place portable shielding that limits radiation exposure through the control room doors and windows will be eliminated.

5.2 Plant Modifications to the VNCR System

The VNCR system current operating modes are described in the FSAR. The VNCR system is classified as non-safety related, however selected components such as the control room charcoal filter fans, the control room recirculation fans, damper controllers and the charcoal/HEPA/roughing filter have been upgraded to augmented quality status. For the accident mode of operation, the two radiation monitor actuation signals are diverse and the radiation monitors are augmented quality status. The containment isolation actuation signals are safety-related and redundant. VNCR Mode 1 is the normal VNCR operating mode.

The AST accident analyses presented in Section 6 of this enclosure assume a new CR emergency mode designated VNCR accident mode. See Figure 5. The VNCR accident mode (Mode 5) provides emergency HEPA/charcoal filtered outside air and HEPA/charcoal filtered recirculating air. To create the VNCR accident mode configuration, the VNCR Mode 4 flow path is modified to include the return air flow path to the emergency fans W14A/B. FPL Energy Point Beach will modify the system to provide redundancy for all active components that are required to reposition from the normal operating position to the accident mode. Auto-start capability will be provided for the CREFS fans, from a diesel-generator supplied source, on loss of offsite power in conjunction with a containment isolation or high control room radiation signal. The diesel generators will remain within the current loading and fuel consumption analysis. The total flow rate is 4950 cfm $\pm$10%. The recirculating flow rate, through the emergency HEPA/charcoal filter unit, is $\geq$ 1955 cfm.

PBNP is proposing to change the normal alignment for the VNCR Mode 1 such that all dampers are aligned to their accident mode position. See Figure 4. Manual damper and fan alignment will still be available in accordance with existing plant operating procedures. (This will require entry of TS 3.7.9, Action Condition A for CREFS inoperability with no automatic reposition function as specified in plant procedures.)

VNCR accident mode operation will slightly change the CRE by the inclusion of the existing ductwork between the CR Return Air Damper (VNCR-4849F) to the CR Charcoal Filter (F-16) that includes damper VNCR-4851B.

The radiological analyses provided in Section 6 of this enclosure assume a total flow rate of 4950 cfm $\pm$10% with $\geq$ 1955 cfm filtered return air.

VNCR-4851B, W-14A/B Return Air Suction Control Damper, is an existing damper. After modification, this damper will open in response to either a containment isolation signal or a high radiation alarm (RE-101 or RE-235) as a part of the new VNCR accident mode configuration. In order to ensure that the system fails to the new VNCR accident mode configuration, existing actuator damper VNCR-04851B will be changed from a Fail Closed to a Fail Open position. The damper will remain capable of manual positioning from the C67 panel. (This will eliminate the current requirement to enter TS 3.7.9, Action Condition A for CREFS inoperability with this damper out of its required position with no automatic reposition function as specified in plant procedures.)

The new VNCR accident mode will be automatically initiated by a containment isolation signal, or by a high radiation signal from the CR area monitor RE-101, or by a high radiation signal from process monitor RE-235 located in the supply duct to the CR. This mode of operation will also be capable of manual alignment from the CR panel C67.

The automatic actuation of VNCR Mode 2 from a containment isolation signal will be removed from the control circuits of the affected dampers. The automatic actuation of VNCR Mode 4 from a radiation monitor signal will also be removed from the control circuits of the affected dampers and fans.

The modifications described above will have no impact on the other, non-radiological functions of the VNCR system. The modifications will include redundancy for all CREFS active components that are required to reposition from their normal operating position and auto-start capability on loss of offsite power in conjunction with a containment isolation or high control room radiation signal from a diesel generator supplied source for the CREFS fans required for the VNCR accident mode. The new VNCR accident mode modifications will be installed and operational (including the necessary plant procedure changes) at the completion of the next refueling outage following NRC approval of the radiological analyses methodology change from TID to AST requested in this LAR. FPL Energy Point Beach is requesting NRC approval of the new VNCR accident mode.

In light of recent industry concerns with regard to CR habitability, initiatives have been taken to further increase system reliability, improve program implementation, gain safety margin, and increase the integrity of the CR 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 CRE with bubble-tight dampers (extremely low leakage dampers) and hard casting the seams of portions of the CRE ductwork.

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 CR washroom exhaust fan; installation of three new bubble tight dampers for the CR, computer room and cable spreading room smoke and heat exhaust fan isolation; upgrades to the CR backup instrument air system; replacement of existing CR washroom exhaust fan with a direct drive fan; and improved differential pressure indication between the CR and the turbine building.

In September 2003, PBNP performed tracer gas testing of the CR in order to determine the amount of unfiltered inleakage for the VNCR Mode 4 configuration that is assumed in the current radiological analyses. In addition to testing the VNCR system in the VNCR Mode 4 configuration, tracer gas testing was also performed for the proposed VNCR accident mode configuration in order to quantify the amount of unfiltered inleakage. The results indicated CR inleakage of 77 ±94 cfm in the proposed VNCR accident mode (Reference 8.25). The unfiltered inleakage rates that are assumed in the proposed radiological analyses presented in Section 6 of this enclosure conservatively bound the maximum measured inleakage rate. Implementation of the proposed modification to the new VNCR accident mode operation is consistent with the September 2003 tracer gas test, and therefore no new tracer gas testing will be performed for post-modification testing.

The remaining inleakage locations are scattered, and the individual leak rates are small. A primary inleakage location was not identified. As such, both inleakage and the fresh air intake assume the same χ/Q values.

5.3 Plant Modifications to the VNPAB System

This LAR proposes an additional deviation from RG 1.183, Section 1.1.2 with respect to manual operator action of the VNPAB system, which is classified as non-safety related. VNPAB fans are powered by safety-related power supplies with diesel generator backup. The system will be modified to provide redundancy for all active components needed to operate the PAB exhaust system. If a LOCA occurs coincident with a LOOP, the VNPAB system will be manually restarted within 30 minutes following the alignment of RHR to containment sump recirculation mode of operation. This will ensure that the auxiliary building vent stack (ABVS) is the source of the release associated with the ECCS leakage phase of the event. This configuration is an input assumption for the LOCA radiological analysis.

The VNPAB system is important to operations, and is therefore required to be highly reliable. For the VNPAB exhaust system to perform its post-accident function of exhausting ECCS leakage from the ABVS, one of two PAB filter fans and one of two PAB stack fans must be able to operate, and the associated exhaust pathway must remain available. The exhaust pathway consists of ductwork, filter housing and a small number of dampers. Each of the two PAB filter fans and each of the two PAB stack fans is powered from a different safety-related diesel generator backed power supply. The filter dampers are designed such that no single failure will block discharge flow.

Once the RHR system is aligned to take pump suction from the containment sump, the operator actions to align the VNPAB system will be accomplished from the CR, with no local operator action required. The controls and instrumentation necessary to place the VNPAB ventilation system in service will be provided in the CR.

New operator actions will be implemented to restore VNPAB operation within 30 minutes following a LOCA with a concurrent LOOP. Restarting the VNPAB ensures that ECCS leakage activity is released to the environment from the PAB vent stack, rather than from other areas of the PAB, with less favorable CR χ/Q values. To support this operation, alternate release locations were studied for the 30-minute restoration time. The results of this study demonstrated that the delay in VNPAB initiation does not adversely impact the CR dose reported at 4.9 Rem TEDE.

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 RG 1.183, 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 RG 1.183 (Reference 8.2). These analyses also include revised CR atmospheric dispersion factors developed using ARCON96 (Reference 8.7). The offsite (EAB and LPZ) and CR dose analyses for the following design basis accident have been reanalyzed using the AST as allowed by 10 CFR 50.67:

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

The FHA, based on the AST, had been submitted previously under Reference 8.4 and approved under Reference 8.5. The revised dose analysis for the fuel handling accident presented in this LAR indicates that the EAB, LPZ and the CR doses meet RG 1.183 acceptance criteria.

For the FHA, higher gap fractions were applied to fuel assuming the limits on burnup and linear heat generation rate are not met, following the method approved by the NRC for Kewaunee Power Station (ML070430020) (Reference 8.6). See Section 3.3 for additional information on gap fractions.

A comparison table of the of the input assumptions for the current, 2007 submittal, and this submittal for each of the accidents analyzed is provided in Appendix A to this enclosure. The input parameters and assumptions for the CLB radiological analyses, the 2007 AST LAR and this LAR are compared in Appendix A of this enclosure. Comparing the assumptions of the CLB to the current LAR finds many differences, most of which are due to the incorporation of the AST methodology and the uprated power level.

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 CR due to airborne radioactivity releases is developed for all of the referenced DBAs. This represents the post-accident dose to the operator due to inhalation and submersion. The CR shielding design is based on the LOCA, which represents the worst case DBA relative to radioactivity releases.

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 piping. 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 CR doses for PBNP uses the following RG 1.183 source term characteristics in place of those identified in TID-14844 (Reference 8.14) and RG 1.4 (Reference 8.23):

- 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 Tables 16 and 17. 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 ECCS leakage into the PAB and into the RWST. No credit for auxiliary building vent stack filtration is taken.

The offsite and CR 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, assuming the most conservative χ/Q. This is determined by calculating the dose during various time intervals. The doses to the LPZ and CR are reported for the duration of the accident (i.e., 30 days). The following sections address topics of significant interest.

Source Term

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 5 lists the nuclides being considered for the LOCA with core melt (nine groups of nuclides). Tables 16 and 17 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.

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. All activity is assumed to be released in the unsprayed volume. The containment volume is 1.0E6 ft^3 with a sprayed fraction of 58.2% of the total (5.82E5 ft^3) (Table 18).

The containment ventilation duct distribution system is designed to promote good mixing of the containment air and ensures that the recirculated cooled air will reach all areas requiring ventilation. The system includes a ring header and branch ducts to the primary compartments for distribution of cooled air from the fan cooler discharge. The cooled air is circulated upward from the lower primary compartments, through the SG compartments to the operating floor level. The ring header discharges air to the containment above the operating floor level. Air that has risen to the containment dome is drawn by the fans through two branch ducts which follow the contour of the containment dome upward on opposite sides of the containment. These ducts take suction at the highest point in the center of the containment. Since all four air handling units discharge into a common ring header, no space in the containment is dependent on a single air handling unit for cooling and ventilation. See Figure 6.

The containment is assumed to leak at the proposed TS leak rate of 0.2 weight % per day (Bases for TS 3.6.1) for the first 24 hours of the accident and then at half that rate (0.1 weight % 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 CS 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 is 60 minutes. The switchover is assumed to take 20 minutes. RHR injection flow will be adjusted in post-modification testing to maintain greater than 500 gpm during the alignment and following initiation of containment spray on recirculation. CS flow on recirculation is greater than 900 gpm. See Figure 3.

In the unlikely event that instrument air is unavailable during recirculation to close the sodium hydroxide (NaOH) additive throttle valves, the spray additive tank discharge valve can be closed manually in the PAB. The manual closure can be performed within the required time and the dose to the operator has been determined to be acceptable.

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 was performed with a recirculation spray duration of 3 hours and 4 hours. The analysis demonstrated that there was insignificant dose impact by reducing the recirculation spray duration time from 4 hours to 3 hours, therefore, the results for the 3-hour duration are reported.

Current FSAR radiological accident analyses do not take credit for operation of the CS system during 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 release from a postulated damaged core with the AST.

New operator actions to align core injection and CS flow to preset throttled positions will be introduced as a result of the new LOCA radiological analysis. Once the RHR system is aligned to take pump suction from the containment sump, all of the manual operator actions to align the RHR and CS systems for recirculation spray will be accomplished from the CR with no local operator action required. All the necessary controls and instrumentation necessary to place the CS system in recirculation spray operation will be provided in the CR.

Extending the effective duration of containment spray has no effect on the analyzed basis for the station's resolution of Generic Letter (GL) 2004-02 (Reference 8.32) concerns with sump screen blockage. The throttling of flow as described above is to keep total flow requirements within the design limits of the RHR pump. The sump screens have been designed and are being tested to demonstrate a flow capacity of 2200 gpm which is slightly in excess of the maximum RHR pump capacity.

Throttling of the CS flow path (flow limiting orifice) and throttling of the RHR core injection flow path (preset throttle position of core deluge injection valves SI-852A/B) will minimize the potential for operator error, while ensuring adequate NPSH is maintained to the running RHR pump, ensuring adequate core injection flow at all times, ensuring GL 2004-02 (Reference 8.32) sump requirements are met, and allowing sufficient time to place recirculation spray in service to meet radiological analyses assumptions associated with the implementation of AST.

The alignment to recirculation spray has been demonstrated on the simulator and the necessary actions were accomplished within 20 minutes.

Alignment to CS on recirculation has no impact on the existing manual operator actions to place RHR on sump recirculation, as the line-up to provide a suction source from RHR to the containment spray for recirculation spray operation occurs after one or both trains of RHR suction have been transferred to the containment sump for the recirculation phase. The CR operator actions to transfer RHR suction from the RWST to the containment sump, and the local operator actions to align cooling to the RHR heat exchanges for the recirculation phase, will remain unchanged.

Containment Sump pH

The containment sump pH varies based on the inputs listed below. An analysis was performed using these inputs that bounds the minimum sump pHs. The analysis concluded that the minimum sump pH at the end of CS chemical addition is between 7 and 8, and accounts for the long-term potential effects of radiolysis of containment contents (including air, water, and chloride bearing electrical cable insulation and jacketing), core inventory spilled to the sump, and accumulations of dry boric acid due to a postulated pre-existing leak.

Containment Sump System Inputs:

- RCS volume: 6389 ft^3
- RCS boron concentration: 2200 ppm
- Safety Injection (SI) accumulator liquid volume: 1136 ft^3 each
- SI accumulator boron concentration: 3100 ppm
- RWST boric acid concentration: 3200 ppm
- Minimum RWST contribution to sump: 226,575 gal
- Maximum RWST measurable volume: 37,901 ft^3
- Spray add tank sodium hydroxide (NaOH) concentration: 30%

Containment Spray Removal of Elemental Iodine

The Standard Review Plan (Reference 8.24) identifies a methodology for the determination of spray removal of elemental iodine independent of the use of spray additive. No credit is taken for removal of elemental iodine by wall deposition. The removal rate constant, for fresh spray, 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. When CS is operating in the recirculation phase, the elemental removal coefficient is reduced to 9.20 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% of the total elemental iodine released to the containment (this is a DF of 200). With the RG 1.183 source term methodology this is considered as being 0.5% of the total inventory of elemental iodine that is released to the containment atmosphere over the duration of gap and in-vessel release phases. In the analysis, this occurs at 2.73 hours.

Containment Spray Removal of Particulates

Particulate spray removal is determined using the model described in Reference 8.24. 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 8.24, 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% 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% 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% 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 3.3 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.

For the analysis, the sedimentation removal coefficient is conservatively assumed to be 0.1 hr^{-1}. This value for sedimentation removal of particulates has been accepted by the NRC for Indian Point Unit 2 (Adams Accession No. ML003727500) (Reference 8.28) and Shearon Harris (Adams Accession No. ML012830516) (Reference 8.29) for the application of the AST methodology. It is assumed that sedimentation removal does not continue beyond a DF of 1000which is reached at 31.6 hours.

ECCS Leakage

When ECCS recirculation is established following a LOCA, leakage is assumed to occur from ECCS equipment outside of containment. 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 modeled in the analysis is 0.21 gpm (i.e., the current analysis value of 400 cc/min total ECCS leakage outside the containment is doubled to 800 cc/min consistent with RG 1.183 guidance). Of the 800 cc/min total ECCS recirculation leakage, 300 cc/min is assumed to leak into the PAB, and 500 cc/min is assumed to leak back to the RWST. The historical data for ECCS leakage, collected from the PBNP Leakage Reduction and Preventive Maintenance Program, are less than the current analysis limit of 400 cc/min. As stated previously, the 400 cc/min has been doubled to 800 cc/min for conservatism in the AST analyses.

Atmospheric dispersion factors appropriate for the calculation of CR doses from these two release paths have been calculated for use in the analysis.

This LAR proposes an additional deviation from RG 1.183, Section 1.1.2 with respect to manual operator action of the VNPAB system, which is classified as non-safety related. VNPAB fans are powered by safety-related power supplies with diesel generator backup. New CR operator actions are required to restore the VNPAB within 30 minutes following the alignment of RHR to containment sump recirculation mode of operation. If a LOCA occurs coincident with a LOOP, the VNPAB system will be manually restarted to ensure that the auxiliary building vent stack is the source of the release associated with the ECCS leakage phase of the event. This configuration is an input assumption for the LOCA radiological analysis.

Once the RHR system is aligned to take pump suction from the containment sump, the operator actions to align the VNPAB system will be accomplished from the CR, with no local operator action required. The controls and instrumentation necessary to place the VNPAB system in service will be provided in the CR.

Procedures will be revised and training for all operating shifts will be completed prior to implementation of the AST license amendment.

Manual restoration of the VNPAB system following the alignment of RHR on sump recirculation has no impact on the existing manual operator actions to place RHR on sump recirculation.

Leakage to the PAB

The analysis models a total ECCS recirculation leakage into the PAB of 300 cc/min beginning at 0 minutes. Ten percent of iodine in the leakage becomes airborne and is released to the outside environment without credit for any retention in the PAB. The dose in the CR from this leakage is 1.94E+00 rem TEDE, to the EAB is 5.08E-01 rem TEDE, and to the LPZ is 3.40E-01 rem TEDE.

Leakage to the RWST

ECCS back-leakage to the RWST is assumed at a rate of 500 cc/min. The iodine in the sump solution is assumed to all be in nonvolatile iodide or iodate form. However, when the solution leaks into the RWST, the iodine will be in an acidic solution, such that there is the possibility of conversion of iodine compounds to form elemental iodine. The amount of iodine that will convert to the elemental form is dependent both on the concentration of iodine in the solution and the pH of the solution. The initial boron concentration in the RWST is conservatively assumed to be 3500 ppm. (This bounds the current TS limit of 3200 ppm.) The initial pH of the RWST solution is determined to be approximately 4.5. The RWST water pH and iodine concentration are determined as a function of time. Figure 3.1 of NUREG-5950 (Reference 8.26) is used to determine the amount of iodine becoming elemental based on the pH and iodine concentration of the RWST solution. With an RWST pH of 4.5 and low iodine concentration, the fraction of conversion to elemental iodine is 2%. By 300 hours, the RWST liquid pH will exceed 5.0 and the indicated conversion to elemental iodine is essentially zero; however, the fraction is conservatively assumed to be 1% for the remainder of the accident duration.

Elemental iodine is volatile and will partition between the liquid and the air in the RWST gas space. The partition factor for elemental iodine is determined to be 45.4 using a relationship to solution temperature from Reference 8.26. This is modeled by the transfer of a portion of the flow to the RWST liquid and a portion to the RWST gas space. The modeling of the air flow out of the RWST is based on a diurnal heating and cooling cycle. This model ignores the effect of the large heat sink provided by the mass of water in the tank that would tend to moderate the effects of the heating and cooling from atmospheric temperature variations. Temperature swings for the RWST are assumed to be within the TS limits and do not result in pressurization of the RWST. The transfer from the RWST gas space to the environment is calculated to be 2.71 cfm based on displacement by the inleakage and air expansion from the heating/cooling cycle. The dose in the CR from this leakage is 4.25E-01 rem TEDE, to the EAB is 1.17E-04 rem TEDE, and to the LPZ is 4.80E-03 rem TEDE.

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

The dose contribution in the CR due to direct shine from the external cloud and from contained sources is addressed. The external cloud contribution includes containment leakage, ECCS leakage, and RWST back-leakage. The contained sources include shine from the containment structure and the CR HVAC filter. The 30-day DDE to a CR operator due to the airborne source in containment, the passing plume source and the CR filter source is calculated.

The analysis takes credit for shielding modifications to the CRE. 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, CR walls and ceiling, etc.).

The external plume is assumed to be uniformly distributed source from El. 26' to approximately 1000 meters above the ground. Similar to the direct shine contribution, only the major intervening shielding is credited (e.g., CR walls and ceiling, turbine building floors etc.).

The new CR VNCR accident mode consists of filtered outside air combined with filtered recirculation. The control room HVAC filter shine dose is calculated based on accumulation of a) the particulate fission products and elemental/organic/particulate iodines resulting from containment leakage and b) elemental/organic iodines resulting from ECCS leakage and RWST back-leakage. The elemental and organic iodines are assumed to be accumulated on the charcoal filter. The particulate iodines and the other particulate fission products are assumed to be accumulated on the HEPA filter. All non-noble gas radioactive materials that enter the control room, either via the control room intake or via control room inleakage, are accumulated on the filter with 100% efficiency. This maximizes the loading on the filter for the duration of the accident.

Computer code SW-QADCGGP is used to calculate the direct shine dose to an operator in the control room from the airborne source inside containment, external plume source, and the control room charcoal/HEPA filter sources. SW-QADCGGP is a Shaw S&W version of the industry standard point-kernel radiation shielding computer code QAD-CGGP. The geometry utilized in the model does not have any significant unaccounted for scattering paths from the source to the receptor. Multiple receptors are placed inside the control room to ensure that the maximum dose is calculated. The source-shield-receptor geometry is such that the dose due to oblique angle scattering is not significant. The most conservative buildup factor is used if the gamma rays traverse through a multiplicity of materials and the last material constitutes less than 3 mean-free-paths.

The dose due to direct shine from the external cloud and contained source is provided 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 VNCR system to switch from the normal operation mode to the VNCR accident mode of operation. The SI setpoint is assumed to be reached immediately at the start of the event and a conservative 60-second delay time for switching from normal to the VNCR accident mode (filtered recirculation with filtered fresh air intake) is modeled.

Acceptance Criteria

The EAB and LPZ dose acceptance criteria for a LOCA are 25 rem TEDE per RG 1.183. This is the 10 CFR 50.67 limit. The acceptance criterion for the CR dose is 5.0 rem TEDE per 10 CFR 50.67. The EAB doses are calculated for the worst 2 hours. The LPZ and CR doses are calculated for 30 days.

Results and Conclusions

The large break LOCA doses in rem TEDE are:

Exclusion Area Boundary	14.2
Low Population Zone	1.6
Control Room – All Pathways (excludes shine)	4.5858
Control Room – Shine	0.28
Control Room – Total Dose	4.9

The EAB dose reported is for the worst 2-hour period, determined to be from 0.5 hours to 2.5 hours.

The acceptance criteria are met.

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 and primary-to-secondary leakage 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 SG tube is assumed to occur. At the start of the accident, radionuclides from the primary coolant enter the SG, 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 prior to reactor trip. The primary-to-secondary break flow results in depressurization of the RCS. Reactor trip and SI are assumed to be automatically initiated simultaneously on low pressurizer pressure. For calculating dose rates, a LOOP is assumed concurrent with the reactor trip; therefore, use of the condenser is lost and the steam is released via the MSSVs or ADVs.

Following reactor trip and SI actuation, 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 the operators have completed the actions necessary to terminate the steam release from the ruptured SG. Pressure between the ruptured SG and the primary system is such that the ruptured SG is not overfilled. 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 analysis assumes 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.

After 30 minutes, steam is assumed to be released only from the intact SG in order to dissipate the core decay heat and to subsequently cool the plant down to the RHR system operating conditions. During post-SGTR cooldown, the pressure in the ruptured SG is assumed to be decreased by the backfill method in which core decay heat and RCS fluid energy are dissipated by releasing steam from the intact SG. This is the preferred approach since it minimizes the radioactivity released to the atmosphere. The plant cooldown to RHR operating conditions is assumed to be accomplished within 30 hours after initiation of the SGTR and steam releases are terminated at this time. A primary and secondary side mass and energy balance is used to calculate the post-trip steam release for the intact SG from 220 seconds to 2 hours, 2 hours to 8 hours, 8 hours to 24 hours, and beyond 24 hours.

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 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 licensed core power. The resulting offsite and CR 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 equilibrium nuclide concentrations in the primary and secondary coolant are discussed in Section 3.2 and presented in Table 6. In addition, two iodine spikes are considered.

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 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 DE I-131. Following the primary system depressurization and reactor trip associated with the SGTR, an iodine spike is initiated in the primary system. The 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. The duration of the spike is 8 hours. See Section 3.4 for additional information on iodine spikes.

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 SG. 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 CR intake. After reactor trip and loss of offsite power, flow to the condenser is isolated.

An iodine partition factor of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) and a particulate retention factor of 0.0025 (based on full power moisture carryover)are applied to both SGs.

The iodine and alkali metal 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.

A portion of the break flow will flash directly to steam upon entering the secondary side of the ruptured SG. There is no radiological analysis penalty taken for tube uncovery in the ruptured SG, hence, the location of the tube rupture is not significant. However, all activity in the flashed break flow is assumed to be transferred out of the SG, and no scrubbing of iodine or particulates, from the flashed break flow, is assumed.

All noble gases in the break flow and primary-to-secondary leakage are assumed to be transferred instantly out of the SG to the atmosphere.

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

Mass Transfer Assumptions

An accident-induced primary-to-secondary leak rate of 1000 gm/min per SG is assumed for the duration of the accident. See Section 2.2.

The integrated tube rupture break flow, break flow flashing fractions, and integrated atmospheric steam releases for the offsite radiological analysis are summarized in Table 19. A range of conditions was addressed as inputs to the calculation of this data.

An auxiliary feedwater flow rate of 200 gpm was analyzed and determined to maintain covered tubes and sufficient primary-to-secondary heat transfer to support the rapid cooldown required for the steam generator tube rupture. The concern was that cooldown through the intact SG PORV may release more mass than can replenished through AFW flow, and as the intact steam generator inventory is reduced, primary-to-secondary heat transfer may be reduced as the tubes uncover. Tube uncovery is also a radiological analysis issue.

The largest contribution to the offsite doses comes from the release of flashed break flow. Flashed break flow, after reactor trip and the assumed LOOP (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 assumptions and methodology for determining the break flow flashing fractions follow. The pre-trip conditions are used to determine the limiting pre-trip flashing fraction. Since flashing is based on the difference between the primary and secondary side fluid enthalpy and the saturation enthalpy on the secondary side, higher flashing is predicted for the case with the higher hot leg temperature. Similarly, the lower secondary side pressure, the greater is the difference in the primary and secondary enthalpies. Although a lower pressure will have a higher heat of vaporization (which would result in less flashing), the lower saturation enthalpy is the determining factor. The primary side fluid enthalpy will not vary greatly with pressure in the range being discussed, but the lowest pressure will produce the highest initial enthalpy.

The highest pre-trip flashing fraction will be predicted with the hot leg temperature of 611.1°F and initial secondary pressure of 601 psia.

Pre-trip Flashing Fraction = 0.22

Prior to trip, the flashed break flow passes through the condenser before the reactor trip, which reduces the iodine concentration in the steam by a factor of 100. With this considered in the analysis, the pre-trip flashing fraction is not significant to the dose, and the conservative, bounding value can be used with a small impact on the results.

The limiting post-trip flashing fraction considers an RCS pressure of 1547 psia and a post-trip SG pressure of 930 psia. The maximum hot leg temperature is 611.1°F. Since flashing is based on the difference between the primary side fluid enthalpy and the saturation enthalpy on the secondary side, higher flashing will be predicted for the case with the higher hot leg temperature. It is conservative to assume that the temperature is not reduced for the

30 minutes in which break flow is calculated. However, with a hot leg temperature of 611.1°F and RCS pressure of 1547 psia, the RCS would be superheated. The hot leg temperature will need to be less than 599.4°F to ensure subcooling of the RCS fluid. Although it is not modeled, it is assumed that the operators will prevent loss of subcooling. As such, the flashing fraction will be calculated assuming saturated conditions in the RCS at 1547 psia.

Post-trip Flashing Fraction = 0.13

Control Room Isolation

The CR VNCR system begins in normal mode. Actuation of the VNCR accident mode is conservatively assumed to occur when the SI/containment isolation actuation setpoint is reached at 220 seconds. 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 one 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 EAB and the 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 CR 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 (30 hours) used in the thermal and hydraulic analysis. The CR doses are calculated for 30 days.

Results and Conclusions

The pre-accident iodine spike doses for the SGTR in rem TEDE are:

Exclusion Area Boundary	2.0
Low Population Zone	0.2
Control Room	1.9

The accident initiated iodine spike doses for the SGTR are:

Exclusion Area Boundary	0.6
Low Population Zone	0.1
Control Room	0.5

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

The acceptance criteria are met.

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 is predicted to occur due to DNB as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed SG 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 ADVs or MSSVs. In addition, iodine and alkali metal 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 SGs following the accident.

Input Parameters and Assumptions

The analysis of the LR radiological consequences uses the analytical methods and assumptions outlined in the RG 1.183, Appendix G. Input parameters and assumptions are provided in Table 20.

Source Term Assumptions

WCAP-16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," has been utilized to estimate the percentage of failed fuel (rods in DNB) assumed in the LR radiological analysis. As a result of this analysis, 30% of the fuel rods in the core are assumed to suffer damage due to DNB, sufficient that all of their gap activity is released to the RCS.

By letter dated April 29, 2004, and as supplemented by letters dated December 16, 2004, and March 22, 2005, Westinghouse Electric Company (Westinghouse) submitted Topical Report (TR) WCAP-16259-P, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," to the NRC for review and approval. The objective of this report was to provide the information and data necessary to allow NRC approval of WCAP-16259-P, Revision 0, as a methodology for a complete nuclear design code system for core design, safety and operational calculations.

The NRC staff reviewed WCAP-16259-P Revision 0, and found it acceptable for referencing in licensing applications for Westinghouse and Combustion Engineering designed pressurized water reactors to the extent specified and under the limitations delineated in the TR. The final NRC SER was issued by letter from Herbert N. Berkow (NRC) to Gresham, J. A. (Westinghouse) "Final Safety Evaluation For Topical Report (TR) WCAP-16259-P," Revision 0, September 15, 2006 (ML052340326) (Reference 8.27). The SER defined the basis for acceptance of the TR.

FPL Energy Point Beach is requesting NRC review and approval of the plant-specific application of WCAP-16259-P-A and associated TS changes to support the LR radiological analysis input assumption that 30% of the fuel rods in the core experience DNB. Enclosure 5 (Proprietary) and Enclosure 6 (Non-Proprietary) provide a WCAP-16259 SER Compliance report for PBNP, demonstrating that no deviation from the NRC SER is proposed. Enclosure 7 provides the SER Compliance Matrices for ANC (SPNOVA) (WCAP-10965-P-A), VIPRE (WCAP-14565-P-A), and RETRAN (WCAP-14882-P-A) and addresses the conditions and limitations associated with the SER for WCAP-10965-P-A, WCAP-14565-P-A and

WCAP-14882-P-A. SPNOVA, VIPRE and RETRAN support WCAP-16259-P. SPNOVA is part of the ANC code which is included in the PBNP CLB.

The rods failing in DNB are high power first or second cycle rods which meet the burnup criteria of RG 1.183, Table 3, Footnote 11. The fuel clad gap activity fractions, applied to the LR, are discussed in Section 3.3 of this enclosure. The activity released from the fuel reflects a radial peaking factor of 1.7.

The equilibrium iodine concentrations in the primary and secondary system are described in Section 3.2 of this enclosure and presented in Table 6.

Dose Calculation Assumptions

An accident-induced primary-to-secondary leak rate of 1000 gm/min per SG is assumed for the duration of the accident. See Section 2.2.

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 which is modeled by a retention factor of 0.0025. This is the estimated full power moisture carryover fraction and is bounding for both units. Because the accident analysis is performed for post-trip low power conditions, this value remains conservative for the analyzed core power of 1811 MWt. These partition and retention factors are applied to the SG primary-to-secondary leakage/steam release pathway. All noble gas activity, transferred to the secondary side of the SG through SG tube leakage, is assumed to be directly released to the outside atmosphere.

For PBNP, plant cooldown to RHR operating conditions can be accomplished within 14 hours after initiation of the locked rotor event and at 30 hours after the accident, the RHR system is assumed to be placed into service, after which 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 SGs from 0 to 2 hours, and hourly from 2 hours to 30 hours. The releases conservatively bound the analyzed core power of 1811 MWt.

Control Room Isolation

The CR HVAC is switched to the VNCR accident mode after receiving a high radiation ventilation system line monitor signal. This signal is reached within one minute, however a conservative time of two minutes was assumed to switch the CR to the VNCR accident mode.

Acceptance Criteria

The EAB and LPZ dose acceptance criterion for a LR is 2.5 rem TEDE per RG 1.183. This is 10% of the 10 CFR 50.67 limit. The CR 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 RHR cut-in time (30 hours). The CR doses are calculated for 30 days.

Results and Conclusions

The LR doses in rem TEDE are:

Exclusion Area Boundary	2.0
Low Population Zone	0.5
Control Room	4.6

The EAB doses reported are for the worst 2-hour period, determined to be from 28 to 30 hours.

The acceptance criteria are met.

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 and alkali metals initially contained in the secondary coolant. In addition, primary coolant activity is transferred via SG tube leaks to the outside atmosphere. A portion of the iodine and alkali metal activity, initially contained in the intact SG, and activity due to tube leakage, is released to the atmosphere through either the ADVs or the 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 CR 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 21.

Source Term Assumptions

The equilibrium nuclide concentrations in the primary and secondary coolant are discussed in Section 3.2 and presented in Table 6. In addition, two iodine spikes are considered.

Pre-accident Spike - A reactor transient has occurred prior to the MSLB and has raised the primary coolant iodine concentration to a conservative value of 60 μCi/gm 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 DE I-131. Following the primary system depressurization and reactor trip associated with the MSLB, an iodine spike is initiated in the primary system. The spike increases the iodine appearance rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to the initial primary system iodine concentration. The duration of the spike is 4 hours. See Section 3.4 for additional information on iodine spikes.

Dose Calculation Assumptions

An accident-induced primary-to-secondary leak rate of 1000 gm/min per SG is assumed for the duration of the accident. See Section 2.2.

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) and a particulate retention factor of 0.0025 (based on full power moisture carryover) are applied to 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.

Within 60 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 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, 2 to 24 hours and 24 to 30 hours. The releases conservatively bound the analyzed core power of 1811 MWt.

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 VNCR system to switch from the normal operation mode to the VNCR accident mode of operation. The analysis conservatively did not credit the SI signal but relied on the ventilation system line radiation monitor signal for CR isolation. It was confirmed that the radiation monitor setpoint is reached within 15 seconds. The VNCR system switches from normal operation to VNCR 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 are 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% of the 10 CFR 50.67 limit. The CR 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 (60 hours). The CR doses are calculated for 30 days.

Results and Conclusions

The MSLB accident doses in rem TEDE are.

For the pre-accident iodine spike:

Exclusion Area Boundary	0.14
Low Population Zone	0.03
Control Room	1.9

For the accident-initiated iodine spike:

Exclusion Area Boundary	0.20
Low Population Zone	0.08
Control Room	4.0

The EAB 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 3.9 to 5.9 hours for the accident initiated iodine spike.

The acceptance criteria are met.

6.5 Control Rod Ejection Accident Doses (CRDE)

For this event, a mechanical failure of a control rod mechanism pressure housing is assumed to have 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 ADVs or the MSSVs. Iodine and alkali metal 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 SGs 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 22.

Source Term

Ten percent of the fuel rods in the core are assumed to suffer sufficient damage due to DNB, as a result of the CRDE, such that all of their gap activity is released to the RCS. The fuel clad gap activity fractions, applied to the CRDE, are discussed in Section 3.3 of this enclosure. Further, 50% of the rods in DNB are assumed to undergo centerline melting, with the melting limited to the inner 10% and occurring over 50% of the axial length, whereby 0.25% of the activity in the core is released as a result of partial melting of the fuel. The fraction of melted fuel activity released to containment or the RCS is 100% for noble gases and 50% for iodines and alkali metals. This is conservative with respect to RG 1.183, Appendix H, Position 1 which states for the containment leakage release path model, the available inventory from the melted fuel is 100% for the noble gases and 25% for the iodines. For the primary-to-secondary leakage model, RG 1.183 specifies 50% for the iodines. Consistent with RG 1.183, the activity releases from the failed/melted fuel reflect the maximum radial peaking factor of 1.7.

The equilibrium nuclide concentrations in the primary and secondary coolant are discussed in Section 3.2 of this enclosure and are presented in Table 6.

Containment Release Pathway

The containment is assumed to leak at the proposed leak rate of 0.2 weight % 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 weight % 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 particulates is credited.

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). For the CRDE analysis, the primary-to-secondary leakage and the steaming from the steam generators is assumed to continue until primary system pressure is less than secondary system pressure. This is assumed to occur at 2000 seconds. This assumption is consistent with previously approved submittals for Indian Point Unit 2 (Adams Accession No. ML003727500) (Reference 8.28), Shearon Harris (Adams Accession No. ML012830516) (Reference 8.29), and Kewaunee Power Station (Adams Accession No. ML070430020) (Reference 8.6). A conservative time of 0.556 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 30 hours. The LR steam releases are conservatively applied for this analysis. The steam releases from a CRDE are conservative since they do not include ECCS injection to absorb decay heat.

As discussed in Section 2.2 of this enclosure, a primary-to-secondary leak rate of 1000 gm/min per SG is assumed for the duration of the accident. Although the primary-to-secondary pressure differential drops throughout the event, the constant flow rate is conservatively maintained.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) and a particulate retention factor of 0.0025 are used.

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

Control Room Isolation

In the event of a CRDE, the low pressurizer pressure SI setpoint will be reached at approximately 76 seconds (rounded up to 90 seconds) after event initiation. The SI/containment isolation signal causes the CR HVAC to switch from the normal operation mode to the post-accident mode of operation. It is assumed that the VNCR system switches from normal operation to VNCR accident mode of operation at 150 seconds (90 seconds for SI/containment isolation signal plus a conservative 60-second delay time).

Acceptance Criteria

The EAB and LPZ dose acceptance criteria for a rod ejection are 6.3 rem TEDE per RG 1.183. This is approximately 25% of the 10 CFR 50.67 limit. The CR 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 and CR doses are calculated for 30 days.

Results and Conclusions

The CRDE doses in rem TEDE are:

Exclusion Area Boundary	2.3
Low Population Zone	0.8
Control Room	2.9

The EAB doses reported are for the worst 2-hour period, determined to be from 0 to 2 hours.

The acceptance criteria are met.

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.

Input Parameters and Assumptions

The input assumptions are presented in Table 23.

This analysis involves dropping a recently discharged fuel assembly. All activity is assumed to be released from the containment refueling cavity or the spent fuel pool to the atmosphere in two hours per the guidance of RG 1.183.

No credit is taken for ventilation filtration system operation in the spent fuel area (i.e., drumming area vent stack). Similarly, no credit is taken for containment purge supply and exhaust system closure or filtration capability. Since the assumptions and parameters used to model the release due to a FHA inside containment are identical to those for a FHA in the spent fuel pool, except for different CR intake atmospheric dispersion factor values (χ/Qs) for the different release paths, the activity released is the same regardless of the location of the accident. In order to bound the accident, the location with the highest χ/Q value is assumed. Therefore, the evaluation presented assumes the accident occurs in the Unit 2 containment building and the release is through the purge stack, resulting in a bounding analysis for a postulated accident in either location. Discussion of the CR and offsite χ/Qs is in Section 4.0 of this enclosure.

The proposed FHA presented in this LAR assumes an increase in core power from 1650 MWt to an analyzed core power of 1811 MWt. The FHA, based on the AST, was submitted previously under Reference 8.4 and approved under Reference 8.5. The previously approved analysis, performed at 1650 MWt, bounds the current licensed power level of 1540 MWt. The revised FHA dose analysis presented in this LAR performed at an analyzed core power of 1811 MWt is intended to support a future EPU license amendment request and is also conservative with respect to the current licensed power level.

Consistent with the current FHA, the dropped assembly is assumed to have been discharged from the core 65 hours after reactor shutdown; therefore, a decay time of 65 hours is applied to the activities in the analysis.

The current FHA assumes a CREFS 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 presented in this LAR, the FHA was re-evaluated assuming a revised CREFS mode that consists of 4950 cfm ±10% total flow and ≥1955 filtered recirculation flow (see Section 5.0 of this enclosure) and a reduced unfiltered inleakage of 300 cfm.

This LAR also used revised atmosphere dispersion factors for the CR (see Section 4.3 of this enclosure).

Source Term Assumptions

In accordance with RG 1.183, the iodine species in the pool is 99.85% elemental and 0.15% organic. This is based on the chemical form of the halogens leaving the fuel to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodine. It assumed that all CsI instantaneously dissociates in the water and re-evolves as elemental.

Consistent with RG 1.183 (Position 1.2 of Appendix B), 100% of the gap activity in the damaged fuel assembly is released. The radionuclides considered for release are xenons, kryptons and iodines. The applicable nuclides and corresponding inventories are provided in Table 23. These values are based on an analyzed core power of 1811 MWt core power. The alkali metals, cesium and rubidium are not included in this analysis because they are not assumed to be released from the pool. Per RG 1.183 (Position 3.0 of Appendix B), the cesium and rubidium (particulate radionuclides) released from the damaged fuel rods are assumed to be retained by the water in the refueling cavity and would not be available for release.

An effective DF of 200 for iodine, as provided in RG 1.183, is used in the analysis to account for scrubbing of the iodine as it evolves through the pool. This DF is applicable to PBNP because the minimum water level requirement of RG 1.183, Appendix B, Step 2 is met. PBNP TS 3.9.6, "Refueling Cavity Water Level," requires that a minimum of 23 feet of water above the top of the reactor vessel flange shall be maintained. Similarly, TS 3.7.10, "Fuel Storage Pool Water Level," requires a minimum of 23 feet of water over the top of the assemblies during movement of irradiated fuel assemblies.

The current FHA analysis assumes gap fractions of 8% for 1-131, 10% for Kr-85 and 5% for all other noble gas and iodine nuclides. This LAR proposes a change for the assumed FHA gap fractions to 12% for I-131, 30% for Kr-85, and 10% for the other iodines and noble gases. This change reflects the assumption that the damaged nuclear fuel used in the FHA radiological analysis is assumed to exceed the RG 1.183, Table 3, Footnote 11 criteria. The gap fractions are those from RG 1.25 with the value for I-131 increased by 20%, consistent with the recommendation in NUREG/CR-5009. This is a conservative assumption and follows the method approved by the NRC for Kewaunee Power Station (Adams Accession No. ML070430020) (Reference 8.6). See Section 3.3.

Control Room Isolation

The current FHA assumes the time to switch to CR isolation and post-accident mode is 10 minutes. For the FHA presented in this LAR, the time to switch the VNCR system from the normal operation to VNCR accident mode is also 10 minutes.

Acceptance Criteria

The EAB and LPZ dose acceptance criterion for the FHA is 6.3 TEDE per RG 1.183. The CR 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). CR doses are calculated for 30 days.

Results and Conclusions

The FHA doses in rem TEDE are:

Exclusion Area Boundary	2.7
Low Population Zone	0.2
Control Room	4.3

The acceptance criteria are 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 predicted.

The initial makeup of the RCS to the vessel is via suction from the RWST to the SI 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.

PBNP Unit 1 and Unit 2 RVHD accident analysis was approved on September 23, 2005 (ML052560089) (Reference 8.30). In the NRC SER, a regulatory commitment that the reactor must be shut down for 100 hours prior to lifting the reactor vessel head is stated. As the RVHD radiological analysis proposed in this LAR assumes the reactor vessel head is lifted immediately following shutdown, the 100 hour regulatory commitment may be eliminated following NRC approval of this LAR.

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 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, interlock is 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.

The input assumptions are presented in Table 24.

Source Term

The accident analysis using the AST methodology 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 fuel melting is assumed to occur.

The 5% gap fraction for iodines for the design basis LOCA is assumed to be applicable to the RVHD. No additional release from the fuel due to melting is assumed. 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. Recirculation is conservatively initiated at 0 minutes. The leakage continues for the 30-day period following the accident.

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 elemental 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 modeled in the analysis is 0.21 gpm (i.e., the current analysis value of 400 cc/min total ECCS leakage outside the containment is doubled to 800 cc/min consistent with RG 1.183 guidance). Of the 800 cc/min total ECCS recirculation leakage, 300 cc/min is assumed to leak into the PAB, and 500 cc/min is assumed to leak back to the RWST. These values are consistent with historical data collected from the PBNP Leakage Reduction and Preventive Maintenance program. Atmospheric dispersion factors appropriate for the calculation of CR doses from these two release paths have been calculated for use in the analysis.

<u>Leakage to the PAB</u>

The analysis models a total ECCS recirculation leakage into the PAB of 300 cc/min beginning at 0 minutes. Ten percent of iodine in the leakage becomes airborne and is released to the outside environment without credit for any retention in the PAB.

<u>Leakage to the RWST</u>

ECCS back-leakage to the RWST is assumed at a rate of 500 cc/min. The iodine in the sump solution is assumed to all be in nonvolatile iodide or iodate form. However, when the solution leaks into the RWST, the iodine will be in an acidic solution such that there is the possibility of conversion of iodine compounds to form elemental iodine. The amount of iodine that will convert to the elemental form is dependent both on the concentration of iodine in the solution and the pH of the solution. The initial boron concentration in the RWST is conservatively assumed to be 3500 ppm. (This bounds the current TS limit of 3200 ppm.) The RWST water pH and iodine concentration are determined as a function of time. Figure 3.1 of NUREG-5950 (Reference 8.26) is used to determine the amount of iodine becoming elemental based on the pH and iodine concentration of the RWST solution. With an RWST pH of 4.5 (conservatively based on the minimum pH in order to obtain the largest possible elemental iodine fraction) and the low iodine concentration that develops during the accident, the estimated fraction of conversion to elemental iodine is 2%, and it is assumed to remain at 2% for the duration of the analysis.

Elemental iodine is volatile and will partition between the liquid and the air in the RWST gas space. The partition factor for elemental iodine is determined to be 45.4 using a relationship to solution temperature from Reference 8.26. Partition is modeled by the transfer of a portion of the flow to the RWST liquid and a portion to the RWST gas space. The modeling of the air flow out of the RWST is based on a diurnal heating and cooling cycle. This model ignores the effect of the large heat sink provided by the mass of water in the tank that would tend to moderate the effects of the heating and cooling from atmospheric temperature variations. The transfer from the RWST gas space to the environment is calculated to be 2.71 cfm based on displacement by the inleakage and air expansion from the heating/cooling cycle.

Control Room Isolation

In the event of a reactor vessel head drop, immediate manual actuation of the VNCR accident mode of operation is assumed to occur. There is sufficient time between accident recognition and release initiation to credit manual actuation of the VNCR accident mode of operation. In addition, the magnitude of the activity released to the environment is large enough to ensure automatic operation of the VNCR accident mode of operation via a high radiation signal. Therefore, no delay in manual actuation of the VNCR accident mode of operation is taken into consideration for the RVHD CR 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 CR 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 CR doses are calculated for the duration of the release (i.e., 30 days).

Results and Conclusions

The RVHD doses in rem TEDE are:

Exclusion Area Boundary	0.1
Low Population Zone	0.1
Control Room	0.5

The EAB doses reported are for the worst 2-hour period, determined to be from 0 to 2 hours.

The acceptance criteria are met.

7.0 Technical Specifications (TS) Impact

Proposed TS changes to support implementation of AST are as follows:

- 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.

- TS Section 3.4.16, RCS Specific Activity

  Surveillance Requirement (SR) 3.4.16.2 is revised to change the specific activity of the reactor coolant from DE I-131 ≤ 0.8 μCi/gm to ≤ 0.5 μCi/gm.

  Note that the current TS Condition C for DE I-131 > 50 μCi/gm is not revised. The AST radiological analyses use a value of 60 μCi/gm which is conservative.

- TS Section 3.7.9, Control Room Emergency Filtration System (CREFS)

  SR 3.7.9.3 is revised to delete the word "makeup" which changes the SR to read "Verify each CREFS emergency fan actuates on an actual or simulated actuation signal."

  SR 3.7.9.6 is revised to delete the word "makeup" twice which changes the SR to read "Verify each CREFS emergency fan can maintain a positive pressure of ≥ 0.125 inches water gauge in the CR envelope, relative to the adjacent turbine building during the emergency mode of operation at a flow rate of 4950 cfm ±10%."

- TS Section 3.7.13, Secondary Specific Activity

  LCO 3.7.13 is revised to change the specific activity of the secondary coolant from ≤ 1.00 μCi/gm to ≤ 0.1 μCi/gm DE I-131.

  SR 3.7.13.1 is revised to change the specific activity of the secondary coolant from ≤ 1.00 μCi/gm to ≤ 0.1 μCi/gm DE I-131.

- TS Section 5.5.15, Containment Leakage Rate Testing Program

  Item c. is revised to change the maximum allowable containment leakage rate, L_a at P_a from 0.4% to 0.2% of containment air weight per day.

- 5.6.4 CORE OPERATING LIMITS REPORT (COLR)

  The list entitled, "The approved analytical methods are described in the following documents: " is revised to include a new item (13).

  (13) WCAP-16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Analyses"

TS changes to support VNCR accident mode are as follows:

Operation in the VNCR accident mode is the assumed mode of operation for the radiological dose analyses presented in Section 6 of this enclosure, and is therefore, the only mode of operation to be addressed by TS Limiting Condition for Operation (LCO) 3.7.9.

TS Section 3.7.9, Control Room Emergency Filtration System (CREFS) Surveillance Requirement (SR) 3.7.9.6 will reflect the new VNCR accident mode operation by deleting the word "makeup" from the surveillance to reflect the change in the airflow distribution to makeup and recirculating airflow. The total filter flow rate of 4950 cfm ±10% will remain the acceptance criteria for the filter test surveillance (TS 5.5.10).

The information provided in the TS Basis 3.7.9 for VNCR modes 1, 2, 3, and 4 will be deleted from the PBNP TS Basis 3.7.9 as this information does not directly describe the design basis accident function of the system, and is duplicative of the information that is currently provided in the PBNP FSAR Section 9.8.

The CR ventilation configuration described in the PBNP FSAR Section 9.8 will be revised to reflect the new VNCR accident mode configuration. The VNCR modes 2 and 4 descriptions will be revised to eliminate the automatic actuation. Manual VNCR damper and fan alignments for all modes of operation will remain procedurally controlled by plant procedures.

Five new commitments for plant modifications and procedure changes are made in support of this application:

- FPL Energy Point Beach will modify the PBNP control room radiation shielding to ensure control room habitability requirements are maintained. This modification is scheduled to be completed following NRC approval of this LAR 241 "Alternative Source Term" at the Unit 1 (2010) refueling outage.

- FPL Energy Point Beach will modify the CS and RHR systems to provide throttling capability of CS and RHR during the ECCS recirculation phase. These modifications will be completed on a unit specific basis at the next Unit 1 (2010) and Unit 2 (2009) refueling outages.

- FPL Energy Point Beach will revise PBNP Emergency Operating Procedures (EOPs) to direct continued CS while on sump recirculation, if containment radiological conditions and/or core damage indicates it is required. These procedure changes will be implemented following NRC approval of this LAR 241 "Alternative Source Term" and following the completion of each unit specific installation of the CS and RHR system modifications to provide throttling capability during the ECCS recirculation phase.

- FPL Energy Point Beach will modify the control room emergency filtration system (CREFS) to create a new alignment for the accident mode that provides a combination of filtered outside air and filtered recirculation air. The modifications will include redundancy for all CREFS active components that must reposition from their normal operating position, and auto-start capability on loss of offsite power in conjunction with a containment isolation or high control room radiation signal from a diesel generator supplied source for the CREFS fans required for the new system alignment. This modification will be completed following NRC approval of this LAR 241 "Alternative Source Term" during the second site refueling outage that completes installation of the CS and RHR system modifications to provide throttling capability during the ECCS recirculation phase, thus completing installation for both units.

- FPL Energy Point Beach will modify the PAB ventilation system (VNPAB) to ensure redundancy of active components needed to operate the PAB exhaust system. VNPAB components credited for AST will be upgraded to an augmented quality status. No credit is taken by AST for the PAB charcoal filters. FPL Energy Point Beach will revise PBNP EOPs to address starting the VNPAB fans.

Applicable TS and Bases markups are provided in Enclosure 2.

References

8.1 "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.

8.2 Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000. (ML003716792)

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

8.4 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. (ML030970703)

8.5 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. (ML040680918)

8.6 NRC Letter to Kewaunee Power Station, "Issuance of Amendment Re: Radiological Accident Analysis and Associated Technical Specifications Change (TAC No. MC9715)," dated March 8, 2007. (ML070430020)

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

8.8 NRC Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)," March 1972.

8.9 NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors" (February 1988).

8.10 WCAP-10965-P-A (Proprietary), "ANC: A Westinghouse Advanced Nodal Computer Code," Y. S. Liu, et al., September 1986.

8.11 WCAP-14565-P-A (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Sung, Y.X., et al., October 1999. (ML993160101)

8.12 WCAP-14882-P-A (Proprietary), "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," Huegel, D.S., et al., April 1999.

8.13 WCAP-16259-P-A (Proprietary), "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," Beard, C.L., et al., August 2006. (ML062580352)

8.14 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.

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

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

8.17 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.

8.18 NRC Regulatory Guide 1.23 (Safety Guide 23), Onsite Meteorological Program, February 17, 1972. (ML020360030)

8.19 US EPA Guidance EPA-454/R-99-005, "Meteorological Monitoring Guidance for Regulatory Modeling Applications," February 2000

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

8.21 Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003. (ML031530505)

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

8.23 Regulatory Guide 1.4, Revision 2, "Assumptions Used For Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactor," June 1974. (ML003739614)

8.24 NUREG-0800, Standard Review Plan 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 4, March 2007. (ML070190178)

8.25 NCS Corporation "Control Room Envelope Inleakage Testing at Point Beach Nuclear Plant – 2003, Final Report," dated January 5, 2004.

8.26 NUREG/CR-5950, "Iodine Evolution and pH Control," E. C Beahm, et al, December 1992.

8.27 Letter from Herbert N. Berkow (NRC) to Gresham, J. A. (Westinghouse) "Final Safety Evaluation For Topical Report (TR) WCAP-16259-P," Revision 0, September 15, 2006 (ML052340326)

8.28 USNRC Letter, “Indian Point Nuclear Generating Unit No. 2 – Re: Issuance of Amendment Affecting Containment Air Filtration, Control Room Air Filtration, and Containment Integrity During Fuel Handling Operations (Tac No. MA6955),” July 27, 2000. (ML003727500)

8.29 USNRC Letter, “Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Re: Steam Generator Replacement and Power Uprate (Tac Nos. MB0199 and MB0782),” October 12, 2001. (ML012830516)

8.30 USNRC Letter, “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 (Tac Nos. MC7650 MC7651),” September 23, 2005. (ML052560089)

8.31 USNRC Letter, “NUREG-0737 Item II.D.3.4 – Control Room Habitability at Point Beach Nuclear Plant Units 1 and 2,” August 10, 1982.

8.32 NRC Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,” September 13, 2004. (ML042360586)

Table 1 Committed Effective Dose Equivalent Dose Conversion Factors			
Isotope	**DCF (Sv/Bq)**	**Isotope**	**DCF (Sv/Bq)**
I-130	7.14E-10	Cs-134	1.25E-08
I-131	8.89E-09	Cs-136	1.98E-09
I-132	1.03E-10	Cs-137	8.63E-09
I-133	1.58E-09	CS-138	2.74E-11
I-134	3.55E-11	Rb-86	1.79E-09
I-135	3.32E-10	Ru-103	2.42E-09
Kr-85m	N/A	Ru-105	1.23E-10
Kr-85	N/A	Ru-106	1.29E-07
Kr-87	N/A	Rh-105	2.58E-10
Kr-88	N/A	Mo-99	1.07E-09
Xe-131m	N/A	Tc-99m	8.80E-12
Xe-133m	N/A	Y-90	2.28E-09
Xe-133	N/A	Y-91	1.32E-08
Xe-135m	N/A	Y-92	2.11E-10
Xe-135	N/A	Y-93	5.82E-10
Xe-138	N/A	Nb-95	1.57E-09
Te-127	8.60E-11	Zr-95	6.39E-09
Te-127m	5.81E-09	Zr-97	1.17E-09
Te-129m	6.47E-09	La-140	1.31E-09
Te-129	2.42E-11	La-142	6.84E-11
Te-131m	1.73E-09	Nd-147	1.85E-09
Te-132	2.55E-09	Pr-143	2.19E-09
Sb-127	1.63E-09	Am-241	1.20E-04
Sb-129	1.74E-10	Cm-242	4.67E-06
Ce-141	2.42E-09	Cm-244	6.70E-05
Ce-143	9.16E-10	Sr-89	1.12E-08
Ce-144	1.01E-07	Sr-90	3.51E-07
Pu-238	1.06E-04	Sr-91	4.49E-10
Pu-239	1.16E-04	Sr-92	2.18E-10
Pu-240	1.16E-04	Ba-139	4.64E-11
Pu-241	2.23E-06	Ba-140	1.01E-09
Np-239	6.78E-10		

Table 2 Effective Dose Equivalent Dose Conversion Factors			
Isotope	DCF ($Sv \cdot m^3/Bq \cdot sec$)	Isotope	DCF ($Sv \cdot m^3/Bq \cdot sec$)
I-130	1.04E-13	Cs-134	7.57E-14
I-131	1.82E-14	Cs-136	1.06E-13
I-132	1.12E-13	Cs-137	2.88E-14
I-133	2.94E-14	CS-138	1.21E-13
I-134	1.30E-13	Rb-86	4.81E-15
I-135	7.98E-14	Ru-103	2.25E-14
Kr-85m	7.48E-15	Ru-105	3.81E-14
Kr-85	1.19E-16	Ru-106	0.0
Kr-87	4.12E-14	Rh-105	3.72E-15
Kr-88	1.02E-13	Mo-99	7.28E-15
Xe-131m	3.89E-16	Tc-99m	5.89E-15
Xe-133m	1.37E-15	Y-90	1.90E-16
Xe-133	1.56E-15	Y-91	2.60E-16
Xe-135m	2.04E-14	Y-92	1.30E-14
Xe-135	1.19E-14	Y-93	4.80E-15
Xe-138	5.77E-14	Nb-95	3.74E-14
Te-127	2.42E-16	Zr-95	3.60E-14
Te-127m	1.47E-16	Zr-97	9.02E-15
Te-129m	1.55E-15	La-140	1.17E-13
Te-129	2.75E-15	La-142	1.44E-13
Te-131m	7.01E-14	Nd-147	6.19E-15
Te-132	1.03E-14	Pr-143	2.10E-17
Sb-127	3.33E-14	Am-241	8.18E-16
Sb-129	7.14E-14	Cm-242	5.69E-18
Ce-141	3.43 E-15	Cm-244	4.91E-18
Ce-143	1.29E-14	Sr-89	7.73E-17
Ce-144	8.53E-16	Sr-90	7.53E-18
Pu-238	4.88E-18	Sr-91	3.45E-14
Pu-239	4.24E-18	Sr-92	6.79E-14
Pu-240	4.75E-18	Ba-139	2.17E-15
Pu-241	7.25E-20	Ba-140	8.58E-15
Np-239	7.69E-15		

Table 3[1] Offsite Breathing Rates and Atmospheric Dispersion Factors	
Time	**Offsite Breathing Rates (m^3/sec)**
0 - 8 hours	3.5E-04[3]
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 CLB.

(2) This EAB atmospheric dispersion factor is conservatively applied during all time intervals in the determination of the limiting 2-hour period.

(3) The offsite breathing rate is held constant for all time intervals for the EAB in the determination of the limiting 2-hour period.

Table 4 **Control Room Parameters**	
Volume	65,243 ft^3
Control Room Unfiltered Inleakage LOCA and MSLB Remaining Non-LOCA events	 200 cfm 300 cfm
Normal Ventilation Flow Rates (VNCR Mode 1) Filtered Makeup Flow Rate Filtered Recirculation Flow Rate Unfiltered Makeup Flow Rate	 0 cfm 0 cfm 2000 cfm
Emergency Mode Flow Rates (VNCR accident mode)[(1)] Filtered Makeup Flow Rate Filtered Recirculation Flow Rate Unfiltered Makeup Flow Rate	 2500 cfm 1955 cfm 0 cfm
Filter Efficiencies Elemental Iodine Organic (Methyl) Iodine Particulate	 95% 95% 99%
Delay to Switch CR HVAC from Normal Operation to Post Accident Operation after receiving an isolation signal (sec)	60 seconds
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

Table 5[1] Core Total Nuclide Activities at Shutdown (Based on 1811 MWt)			
Isotope	**Activity (Ci)**	**Isotope**	**Activity (Ci)**
GROUP 1 - Noble Gases		**GROUP 6 - Barium**	
Kr-85	6.15E+05	Ba-139	9.42E+07
Kr-85m	1.36E+07	Ba-140	9.05E+07
Kr-87	2.68E+07		
Kr-88	3.60E+07	**GROUP 7 - Noble Metals**	
Xe-131m	5.55E+05	Ru-103	7.79E+07
Xe-133	1.02E+08	Ru-105	5.42E+07
Xe-133m	3.21E+06	Ru-106	2.54E+07
Xe-135	2.17E+07	Rh-105	5.08E+07
Xe-135m	2.20E+07	Tc-99m	8.47E+07
Xe-138	9.05E+07	Mo-99	9.62E+07
GROUP 2 - Halogens		**GROUP 8 - Cerium**	
I-130	1.05E+06	Ce-141	8.52E+07
I-131	5.10E+07	Ce-143	8.03E+07
I-132	7.47E+07	Ce-144	6.72E+07
I-133	1.06E+08	Pu-238	1.33E+05
I-134	1.19E+08	Pu-239	1.45E+04
I-135	1.01E+08	Pu-240	2.25E+04
		Pu-241	5.73E+06
GROUP 3 - Alkali Metals (Rb / Cs)		Np-239	9.65E+08
Rb-86	9.95E+04		
Cs-134	9.52E+06	**GROUP 9 - Lanthanides**	
Cs-136	2.14E+06	Y-90	5.01E+06
Cs-137	6.27E+06	Y-91	6.56E+07
Cs-138	9.89E+07	Y-92	6.82E+07
		Y-93	7.67E+07
GROUP 4 - Tellurium		Nb-95	8.87E+07
Te-127	4.54E+06	Zr-95	8.76E+07
Te-127m	7.48E+05	Zr-97	8.80E+07
Te-129	1.33E+07	La-140	9.69E+07
Te-129m	2.52E+06	La-142	8.25E+07
Te-131m	9.95E+06	Pr-143	7.75E+07
Te-132	7.30E+07	Nd-147	3.33E+07
Sb-127	4.63E+06	Am-241	6.16E+03
Sb-129	1.42E+07	Cm-242	1.70E+06
		Cm-244	1.58E+05
GROUP 5 - Strontium			
Sr-89	5.03E+07		
Sr-90	4.80E+06		
Sr-91	6.30E+07		
Sr-92	6.73E+07		

Notes

(1) See FSAR Table 14.3.5-1 for LOCA CLB core activities, FSAR Table 14.1.8-4 for LR, CRDE CLB core activities.

Table 6
Reactor and Secondary Coolant Nuclide Concentrations

RCS Nuclide Concentrations (µCi/gm)

Nuclide	All Nuclides 1% defects	Iodine 0.5 µCi/gm DE I-131	Iodine 60 µCi/gm DE I-131	Noble Gas 520 µCi/gm DE Xe-133	Alkali Metal Consistent with 0.5 µCi/gm DE I-131
I-130	2.16E-02	2.89E-03	3.47E-01		
I-131	2.82E+00	3.77E-01	4.52E+01		
I-132	3.17E+00	4.24E-01	5.09E+01		
I-133	4.90E+00	6.55E-01	7.86E+01		
I-134	7.46E-01	9.97E-02	1.20E+01		
I-135	2.81E+00	3.76E-01	4.51E+01		
Kr-85m	2.17E+00			1.58E+00	
Kr-85	1.05E+01			7.63E+00	
Kr-87	1.44E+00			1.05E+00	
Kr-88	4.01E+00			2.92E+00	
Xe-131m	3.23E+00			2.35E+00	
Xe-133m	5.23E+00			3.80E+00	
Xe-133	2.91E+02			2.12E+02	
Xe-135m	5.96E-01			4.33E-01	
Xe-135	9.25E+00			6.72E+00	
Xe-138	7.94E-01			5.77E-01	
Cs-134	2.46E+00				3.29E-01
Cs-136	2.57E+00				3.44E-01
Cs-137	2.09E+00				2.79E-01
Cs-138	1.21E+00				1.62E-01
Rb-86	2.72E-02				3.64E-03

Secondary Coolant Nuclide Concentrations (µCi/gm)

Iodine		Alkali Metal	
Nuclide	**0.1 µCi/gm DE I-131**	**Nuclide**	**Consistent with 0.1 µCi/gm DE I-131**
I-130	5.78E-04	Cs-134	6.58E-02
I-131	7.54E-02	Cs-136	6.88E-02
I-132	8.48E-02	Cs-137	5.58E-02
I-133	1.31E-01	Cs-138	3.42E-02
I-134	1.99E-02	Rb-86	7.28E-04
I-135	7.52E-02		

Table 7 Nuclide Half Life			
Nuclide	**Half Life (sec)**	**Nuclide**	**Half Life (sec)**
Kr-85m	1.61E+04	Sr-92	9.76E+03
Kr-85	3.38E+08	Ba-139	4.96E+03
Kr-87	4.58E+03	Ba-140	1.10E+06
Kr-88	1.02E+04	Ru-103	3.39E+06
Xe-131m	1.03E+06	Ru-105	1.60E+04
Xe-133m	1.89E+05	Ru-106	3.18E+07
Xe-133	4.53E+05	Rh-105	1.27E+05
Xe-135m	9.17E+02	Mo-99	2.38E+05
Xe-135	3.27E+04	Tc-99m	2.17E+04
Xe-138	8.50E+02	Ce-141	2.81 E+06
I-130	4.45E+04	Ce-143	1.19E+05
I-131	6.95E+05	Ce-144	2.46E+07
I-132	8.28E+03	Pu-238	2.77E+09
I-133	7.49E+04	Pu-239	7.59E+11
I-134	3.16E+03	Pu-240	2.06E+11
I-135	2.38E+04	Pu-241	4.54E+08
Cs-134	6.50E+07	Np-239	2.03E+05
Cs-136	1.13E+06	Y-90	2.30E+05
Cs-137	9.46E+08	Y-91	5.05E+06
Cs-138	1.93E+03	Y-92	1.27E+04
Rb-86	1.61 E+06	Y-93	3.64E+04
Te-127m	9.42E+06	Nb-95	3.04E+06
Te-127	3.37E+04	Zr-95	5.53E+06
Te-129m	2.90E+06	Zr-97	6.08E+04
Te-129	4.18E+03	La-140	1.45E+05
Te-131 m	1.08E+05	La-142	5.55E+03
Te-132	2.82E+05	Nd-147	9.49E+05
Sb-127	3.33E+05	Pr-143	1.17E+06
Sb-129	1.56E+04	Am-241	1.36E+10
Sr-89	4.36E+06	Cm-242	1.41 E+07
Sr-90	9.18E+08	Cm-244	5.71 E+08
Sr-91	3.42E+04		

Table 8 ARCON96 Input - Unit 2 Containment	
Input Parameter	**Value**
Meteorological Data	PB2000.met
ARCON96 Wind Speed Units	Miles per hour (mph)
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 ARCON96 Input - Auxiliary Building Vent Stack (ABVS)	
Input Parameter	**Value**
Meteorological Data	PB2000.met
ARCON96 Wind Speed Units	Miles per hour (mph)
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 ARCON96 Input - Unit 2 "A" Main Steam Safety Valves (MSSVs)	
Input Parameter	**Value**
Meteorological Data	PB2000.met
ARCON96 Wind Speed Units	Miles per hour (mph)
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 ARCON96 Input - Unit 2 "B" Main Steam Safety Valves (MSSVs)	
Input Parameter	**Value**
Meteorological Data	PB2000.met
ARCON96 Wind Speed Units	Miles per hour (mph)
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 ARCON96 Input - Unit 2 Containment Façade Penetration	
Input Parameter	**Value**
Meteorological Data	PB2000.met
ARCON96 Wind Speed Units	Miles per hour (mph)
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 ARCON96 Input - Unit 2 Purge Stack	
Input Parameter	**Value**
Meteorological Data	PB2000.met
ARCON96 Wind Speed Units	Miles per hour (mph)
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 ARCON96 Input - Unit 1 Refueling Water Storage Tank (RWST)	
Input Parameter	**Value**
Meteorological Data	PB2000.met
ARCON96 Wind Speed Units	Miles per hour (mph)
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	45 m
Release Type	Ground
Release Height	16.4 m
Building Area	2377 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	195°
Wind Direction Sector Width	90°
Distance to Control Room Air Intake	37 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 15 ARCON96 Input - Unit 2 Refueling Water Storage Tank (RWST)	
Input Parameter	**Value**
Meteorological Data	PB2000.met
ARCON96 Wind Speed Units	Miles per hour (mph)
Height of Lower Wind Speed Instrument	10 m
Height of Upper Wind Speed Instrument	45 m
Release Type	Ground
Release Height	16.4 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	24.3 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 16 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 17 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 18[1] **Assumptions Used for Large Break LOCA Dose Analysis**	
Core Activity	See Table 5
Activity Release Fractions and Timing	See Tables 16 & 17
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
Spray Operation Injection Sprays Initiated Injection Sprays Terminated Delay Time to Recirculation Sprays Recirculation Spray Duration	 0 – 90 seconds 1.00 hour 20 minutes 3 hours
Average Spray Fall Height	65.58 feet
Spray Flow Rates Injection Recirculation	 1,070 gpm 900
Containment Spray Removal Coefficients Spray elemental iodine removal Injection Recirculation Spray particulate removal Injection Recirculation	 20.0 hr^{-1} 9.20 4.42 3.72
Containment Spray DF Elemental Particulate	 200 1000[2]
Sedimentation Particulate Removal (Unsprayed region: From start of event; Sprayed region: When sprays not operating.)	0.1 hr^{-1}
Containment Sump Volume	2.43E+05 gal
Containment Sump pH	≥ 7.0

Notes:

(1) CLB information for LOCA is contained in PBNP FSAR 14.3.5.

(1) Due to spray and sedimentation removal.

Table 18 (Cont.) **Assumptions Used for Large Break LOCA Dose Analysis**	
RWST Minimum Water Volume	25,500 gal
RWST Maximum Air Volume	270,000 gal
RWST Minimum Temperature	40° F
RWST Maximum Temperature	100° F
RWST Maximum Boron Concentration	3500 ppm(6)
Time to Initiate ECCS Recirculation	0 min
ECCS Leak Rate PAB Leak Rate RWST Leak Rate	800 cc/min 300 500
ECCS Leakage Iodine Airborne Fraction PAB Fraction RWST Fraction	 10.0% See Section 6.1
Iodine Species ECCS Leakage Released to the Atmosphere Elemental Organic	 97% 3%
Atmospheric Dispersion (χ/Q) Factors Control Room, Containment Surface(3): 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 1.39E-03 sec/m^3 9.80E-04 3.84E-04 3.46E-04 3.02E-04
Atmospheric Dispersion (χ/Q) Factors Control Room, Auxiliary Building Vent Stack(4): 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 1.80E-03 sec/m^3 1.31E-03 5.15E-04 4.03E-04 3.03E-04
Atmospheric Dispersion (χ/Q) Factors Control Room, Unit 2 RWST(5): 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 9.89E-03 sec/m^3 7.98E-03 2.88E-03 2.75E-03 2.35E-03

Notes:

(2) Used for activity released via containment leakage

(3) Used for activity released via ECCS leakage to Auxiliary Building

(4) Used for activity released via ECCS leakage to RWST.

(5) Sump pH calculation assumes maximum boron value of 3200 ppm

Table 19[1] **Assumptions Used for SGTR Dose Analysis**		
Source Data		
Reactor Coolant Iodine Activity (Initial) Pre-Accident Spike Accident-Initiated Spike Noble Gas Alkali Metal	See Table 6. 60 µCi/gm DE I-131 0.5 µCi/gm DE I-131 520 µCi/gm DE Xe-133 Corresponds to 0.5 µCi/gm DE I-131	
Reactor Coolant Accident-Initiated Iodine Appearance Rate Spike Factor	See Section 3.4 335 times equilibrium rate	
Primary-to-Secondary Leak Rate	1000 gm/min per SG	
Duration of Accident-Initiated Iodine Spike	8 hrs.	
Secondary Coolant Activity (Initial) Iodine Alkali Metal	See Table 6 0.1 µCi/gm DE I-131 Corresponds to 0.1 µCi/gm DE-I-131	
Release Modeling		
Reactor Coolant Initial Mass	1.06E+08 gm	
Steam Generator Initial Mass (each)	2.99E7 gm/SG	
Offsite power	Lost at time of reactor trip (220 sec)	
Primary-to-Secondary Leakage Duration for intact SG	30 hours	
Activity Release Data		
Ruptured Steam Generator		
Pre-trip Break Flow	21,300 lbm	(0 – 220 sec)
Post-trip Break Flow	103,200 lbm	(220 sec - 30 min)
Pre-trip Flashed Break Flow	4,690 lbm	(0 – 220 sec)
Post Trip Flashed Break Flow	13,420 lbm	(220 sec - 30 min)
Steam Release	1130 lbm/sec 88,100 lbm	(0 – 220 sec) (220 sec - 30 min)
SG Iodine Partition Factor: Non-flashed Flashed	 0.01 1.0	
SG Particulate Retention Non-flashed Flashed	 0.0025 1.0	
Condenser Partition Factor	0.01	

Notes:

(1) CLB information for SGTR is contained in PBNP FSAR 14.2.4

Table 19 (Cont.) **Assumptions Used for SGTR Dose Analysis**		
Intact Steam Generator		
Primary-to-Secondary Leakage	1000 gm/min per SG	
Steam Release	1130 lbm/sec	(0 – 220 sec)
	257,700 lbm	(220 sec – 2 hr)
	584,000 lbm	(2 – 8 hr)
	866,000 lbm	(8 – 24 hr)
	54,100 lbm/hr	(>24 hr)
SG Iodine Partition Factor	0.01	
SG Particulate Retention	0.0025	
Condenser Partition Factor	0.01	
Iodine Species Released to the Atmosphere		
Elemental	97%	
Organic	3%	
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	
8 – 24 hours	1.17E-03	
24 – 96 hours	1.07E-03	
96 – 720 hours	9.05E-04	

Table 20[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	30% of core
Gap Fractions I-131 Kr-85 Other Iodines and Noble Gases Alkali Metals	See Section 3.3. 0.08 0.10 0.05 0.12
Radial Peaking Factor	1.7
Reactor Coolant Activity (Initial) Iodine Noble Gas Alkali Metal	See Table 6 0.5 μCi/gm DE I-131 520 μCi/gm DE Xe-133 Corresponds to 0.5 μCi/gm DE I-131
Secondary Coolant Activity (Initial) Iodine| Alkali Metal	See Table 6. 0.1 μCi/gm DE I-131 Corresponds to 0.1 μCi/gm DE I-131
Release Modeling	
Primary-to-Secondary Leakage	2000 gm/min total
Steam Release to Environment	See Table 20A below
SG Iodine Partition Factor	0.01
SG Alkali Metal Retention Factor	0.0025
Iodine Species Released to the Atmosphere Elemental Organic	 97% 3%
RCS Mass	1.06E8 gm
Secondary Side mass 0 – 2 hours > 2 hours	 5.98E+07 gm total 7.37E+07 gm total
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 1.17E-03 1.07E-03 9.05E-04

Notes:

(1) CLB information for LR is contained in PBNP FSAR 14.1.8.

Table 20A Locked Rotor Dose Analysis Steam Release to Environment			
Hours	**Mass (lbm)**	**Hours**	**Mass (lbm)**
0-2	213,295	16-17	37,245
2-3	75,645	17-18	36,821
3-4	70,441	18-19	36,116
4-5	65,672	19-20	35,461
5-6	63,060	20-21	34,969
6-7	60,838	21-22	34,969
7-8	58,305	22-23	34,109
8-9	57,015	23-24	33,595
9-10	55,886	24-25	33,595
10-11	54,629	25-26	33,595
11-12	53,326	26-27	33,595
12-13	52,514	27-28	33,595
13-14	51,714	28-29	33,595
14-15	38,508	29-30	33,595
15-16	37,749		

Table 21[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 Alkali Metal	See Table 6 60 μCi/gm DE I-131 0.5 μCi/gm DE I-131 520 μCi/gm DE Xe-133 Corresponds to 0.5 μCi/gm DE I-131
Reactor Coolant Accident-Initiated Iodine Appearance Rate Spike Factor	See Section 3.4 500 times equilibrium rate
Duration of Accident-Initiated Iodine Spike	4 hours
Secondary Coolant Activity (Initial) Iodine Alkali Metal	See Table 6 0.1 μCi/gm DE I-131 Corresponding to 0.1 μCi/gm DE I-131
Release Modeling	
Primary-to-Secondary Leakage	1000 gm/min per SG
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)	60 hrs
Iodine Form (Atmospheric Release) Elemental Organic	 97% 3%
Steam Releases to Environment 0 – 2 hours 2 – 24 hours 24 - 30 hours	 221,153 lbm 1,048,064 201,570
SG Iodine Partition Factor Faulted SG Intact SG	 1.0 0.01
SG Particulate Retention Factor	0.0025
RCS Mass	1.06E+08 gm
Initial Intact SG mass	2.99E+07 gm

Notes:
(1) CLB information for MSLB is contained in PBNP FSAR 14.2.5.

Table 21 (Cont.) **Assumptions Used for Steam Line Break Dose Analysis**	
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 1.17E-03 1.07E-03 9.05E-04
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 5.11E-03 4.94E-03 4.23E-03

Notes:

(2) Used for activity released via the intact steam generator.

(3) Used for activity released via the faulted steam generator.

Table 22[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	 0.10 0.10 0.12
Fraction of Fuel Melting	0.25% of core
Radial Peaking Factor	1.7
Fraction of Activity Released from Melted Fuel Containment Leakage Iodine Noble Gas Alkali Metals Primary-to-Secondary leakage Iodine Noble Gas Alkali Metals	 50% 100% 50% 50% 100% 50%
Reactor Coolant Activity (Initial) Iodine Noble Gas Alkali Metal	See Table 6. 0.5 μCi/gm DE I-131 520 μCi/gm DE Xe-133 Corresponds to 0.5 μCi/gm DE I-131
Secondary Coolant Activity (Initial) Iodine Alkali Metal	See Table 6. 0.1 μCi/gm DE I-131 Corresponding to 0.1 μCi/gm DE I-131
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
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}

Table 22 (Cont.) Assumptions Used for Rod Ejection Dose Analysis	
Primary-to-Secondary Leakage Release Path	
Primary-to-Secondary Leakage Leakage Rate Duration	 2000 gm/min total 2000 sec
Steam Release to Environment 0 – 2 hours 2 – 14 hours 14 – 30 hours	 213,295 lbm 719,045 561,112
SG Iodine Partition Coefficient	0.01
SG Alkali Metal Retention Factor	0.0025
Iodine Chemical Form After Release to Atmosphere Elemental Organic	 97% 3%
RCS Mass	1.06E+08 gm
Total SG Mass	5.98E+07 gm
Atmospheric Dispersion (χ/Q) Factors Control Room, Unit 2 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 3.84E-04 3.46E-04 3.02E-04
Atmospheric Dispersion (χ/Q) Factors Control Room, Unit 2 "A" Safeties(3): 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 4.66E-03 sec/m^3 3.40E-03 1.17E-03 1.07E-03 9.05E-04

Notes:

(1) CLB information for CRDE is contained in PBNP FSAR 14.2.6.

(2) Used for activity released via containment leakage.

(3) Used for activity released via secondary releases.

Table 23[1] Assumptions Used for FHA in Containment Dose Analysis	
Radial Peaking Factor	1.7
Fuel Damaged	1 assembly
Time from Shutdown before Fuel Movement	65 hrs
Activity Released from Water Pool	
I-130	1.95E-01 Ci
I-131	3.50E+02
I-132	2.97E+02
I-133	8.78E+01
I-135	7.45E-01
Kr-85m	8.32E-01
Kr-85	2.59E+03
Kr-87	1.59E-11
Kr-88	6.53E-03
Xe-131m	7.69E+02
Xe-133m	2.81E+03
Xe-133	1.18E+05
Xe-135m	2.43E+01
Xe-135	2.46E+03
Iodine chemical form in pool	
Elemental	99.85%
Organic (methyl)	0.15%
Gap Fractions	
I-131	0.12
Kr-85	0.30
Other Iodines and Noble Gases	0.10
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
Delay to Switch CR HVAC from Normal Operation to Post Accident Operation (VNCR accident mode) After Receiving an Isolation Signal	10 minutes
Atmospheric Dispersion (χ/Q) Factors	
Control Room, Unit 2 Purge Stack[2]:	
0 – 2 hours	6.94E-03 sec/m^3

Note:

(1) CLB 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 24[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	0.05
Iodine Form Released to Environment Elemental Organic	 100% 0%
Recirculation Initiation Time	Immediate
ECCS Leak Rate to Auxiliary Building	300 cc/min
ECCS Leak Rate to RWST	500 cc/min
Iodine Airborne Fraction Leakage to PAB Leakage to RWST	 10% See Section 6.7
Containment Sump Volume	2.43E+05 gal
RWST Minimum Water Volume	25,500 gal
RWST Maximum Air Volume	270,000 gal
RWST Minimum Temperature	40° F
RWST Maximum Temperature	100° F
RWST Maximum Boron Concentration	3500 ppm
Control Room Isolation	Immediate Manual Operator Action
Atmospheric Dispersion (χ/Q) Factors Control Room, Auxiliary Building Vent Stack[2]: 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 1.80E-03 sec/m^3 1.31E-03 5.15E-04 4.03E-04 3.03E-04
Atmospheric Dispersion (χ/Q) Factors Control Room, RWST 0 – 2 hours 2 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 9.89E-03 sec/m^3 7.98E-03 2.88E-03 2.75E-03 2.35E-03

Note:

(1) CLB information for RVHD is contained in PBNP FSAR 14.3.6.

(2) Used for activity released via ECCS leakage to Auxiliary Building.

Table 25 Acronyms			
ADV	Atmospheric dump valve	KI	Potassium iodide
AST	Alternative source term	LAR	License amendment request
CLB	Current licensing basis	LOCA	Loss of coolant accident
CR	Control room	LOOP	Loss of offsite power
CRDE	Rupture of a control rod drive mechanism housing	LPZ	Low population zone
CRE	Control room envelope	LR	Locked rotor
CEDE	Committed effective dose equivalent	MSLB	Main steam line break
CREFS	Control room emergency filtration system	MSSV	Main steam safety valve
CS	Containment spray	NRC	Nuclear Regulatory Commission
DBA	Design basis accident	PAB	Primary auxiliary building
DCF	Dose conversion factor	PBNP	Point Beach Nuclear Plant
DDE	Deep dose equivalent	RCS	Reactor coolant system
DE	Dose equivalent	RG	Regulatory guide
DF	Decontamination factor	RHR	Residual heat removal
DNB	Departure from nucleate boiling	RVHD	Reactor vessel head drop
EAB	Exclusion area boundary	RWST	Refueling water storage tank
ECCS	Emergency core cooling system	SER	Safety evaluation report
EDE	Effective dose equivalent	SG	Steam generator
EPA	Environmental Protection Agency	SGTR	Steam generator tube rupture
EPU	Extended power uprate	SI	Safety injection
EQ	Environmental qualification	TEDE	Total effective dose equivalent
FHA	Fuel handling accident	TID	Technical Information Document
FSAR	Final safety analysis report	TR	Topical report
GDC	General design criteria	TS	Technical specification
GL	Generic letter	VNCR	Control room ventilation system
		VNPAB	Primary auxiliary building ventilation

Figure 1
Unit 1, Unit 2 and Common Release Locations

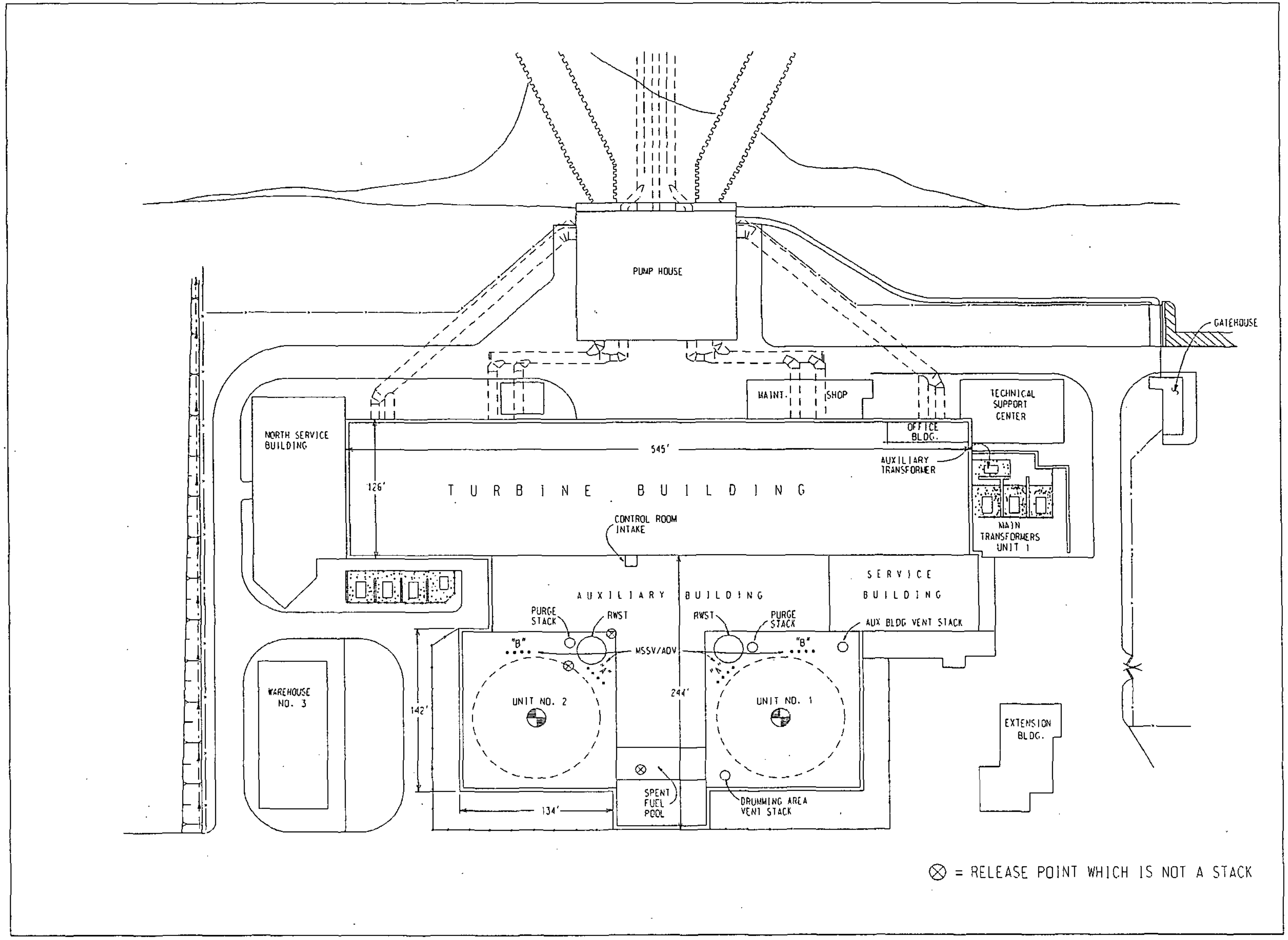

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Figure 2
Point Beach Nuclear Plant Site Plan

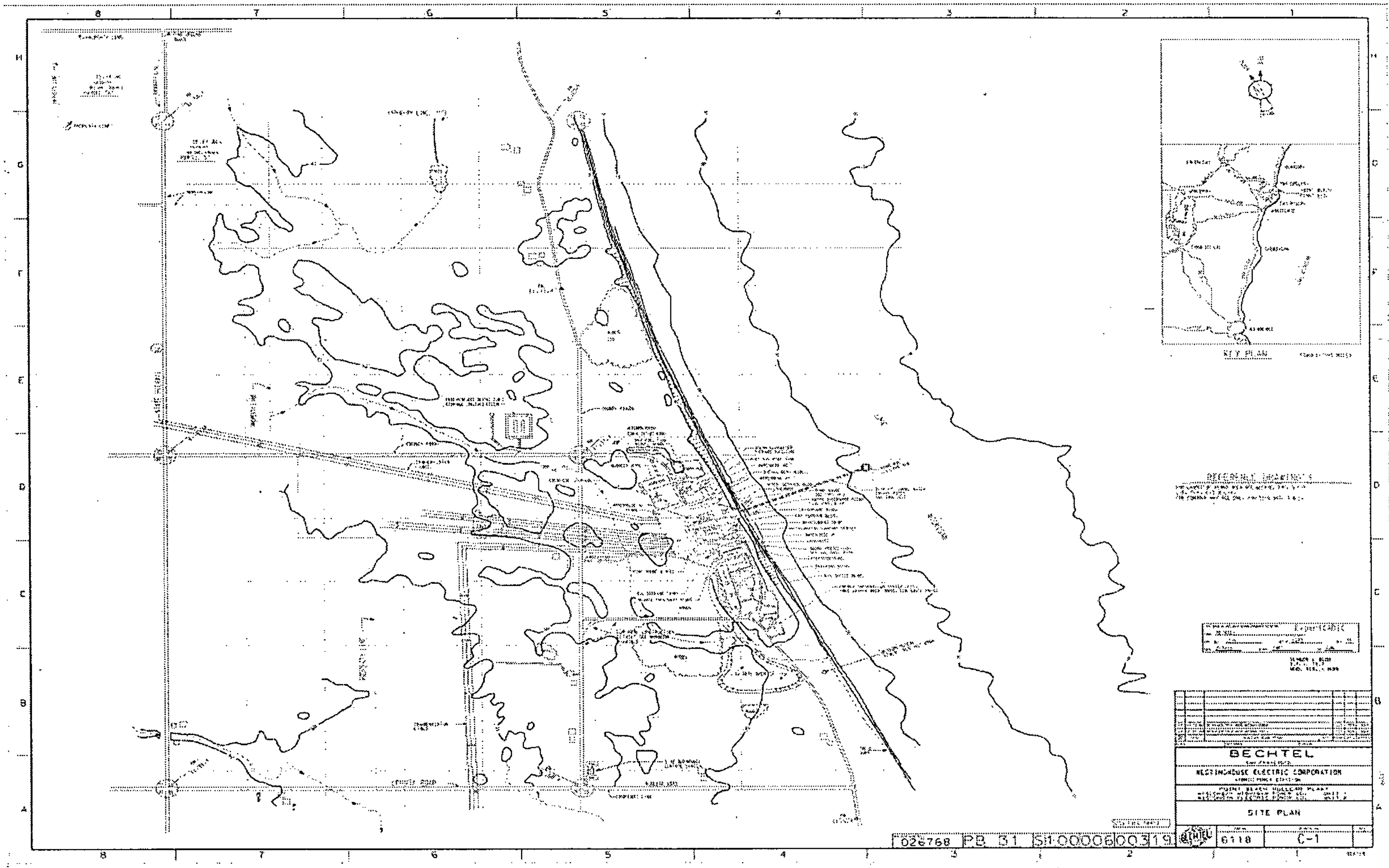


Figure 3
Containment Spray on Recirculation

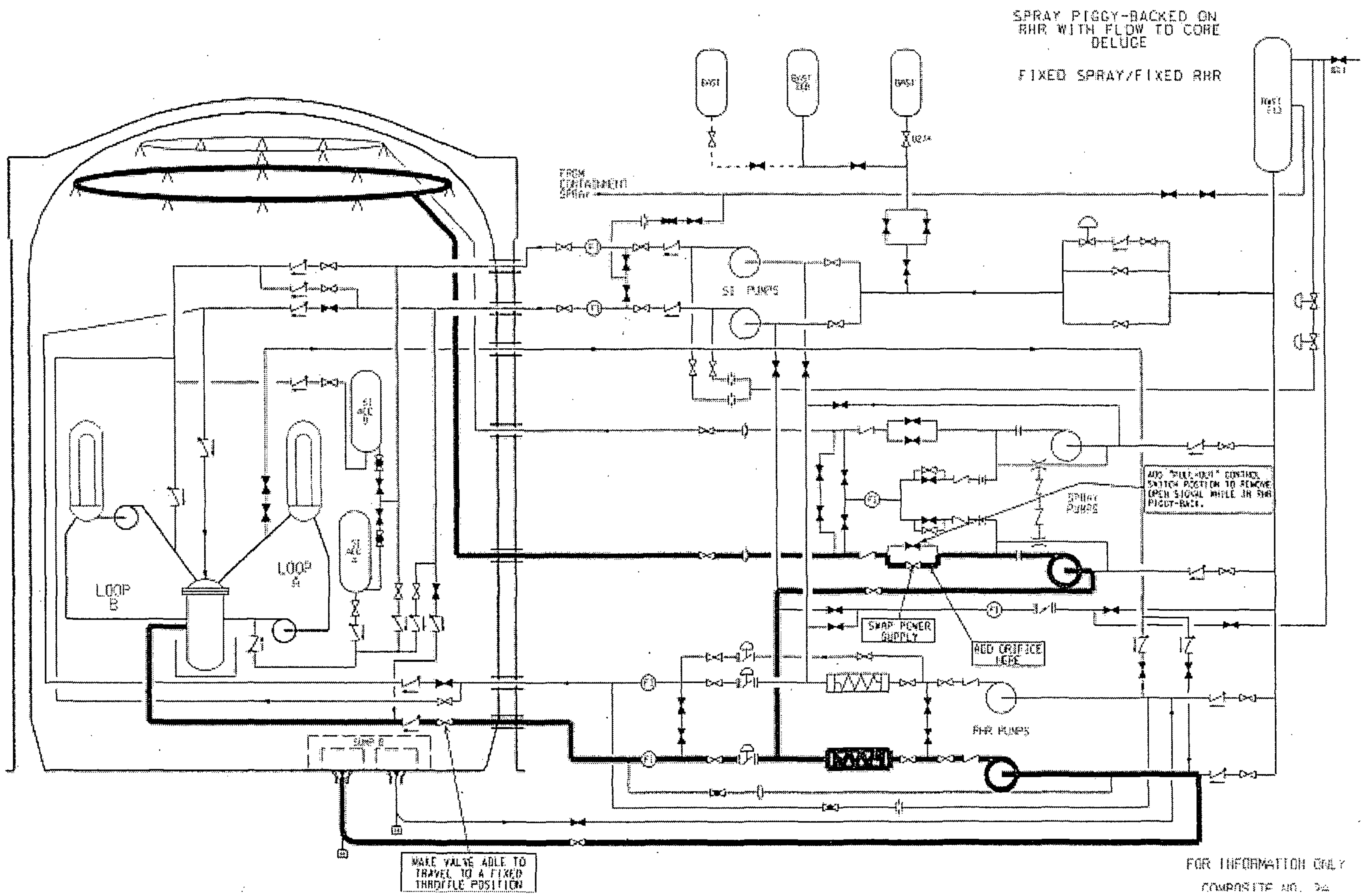
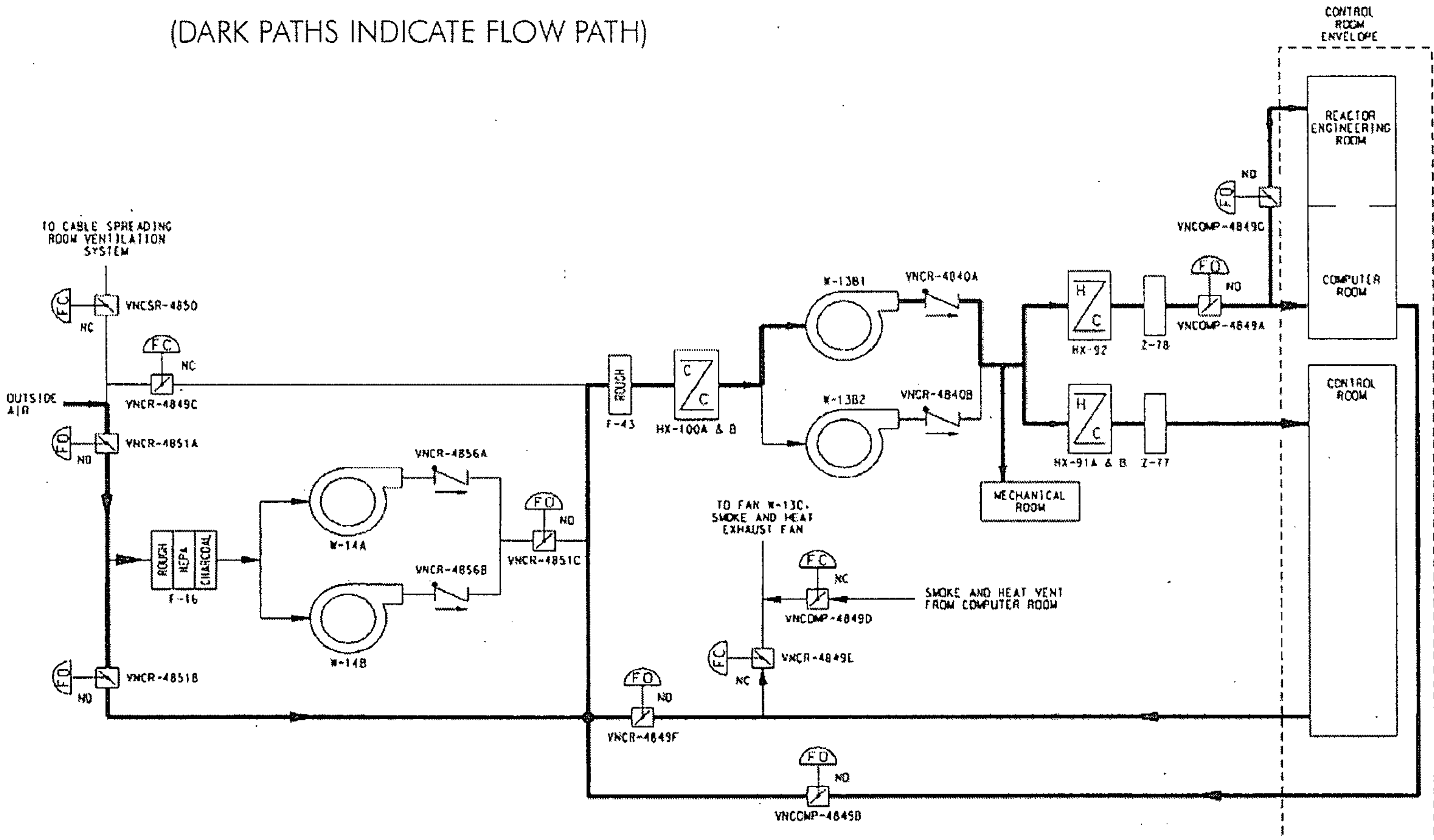


Figure 4

PROPOSED NORMAL MODE

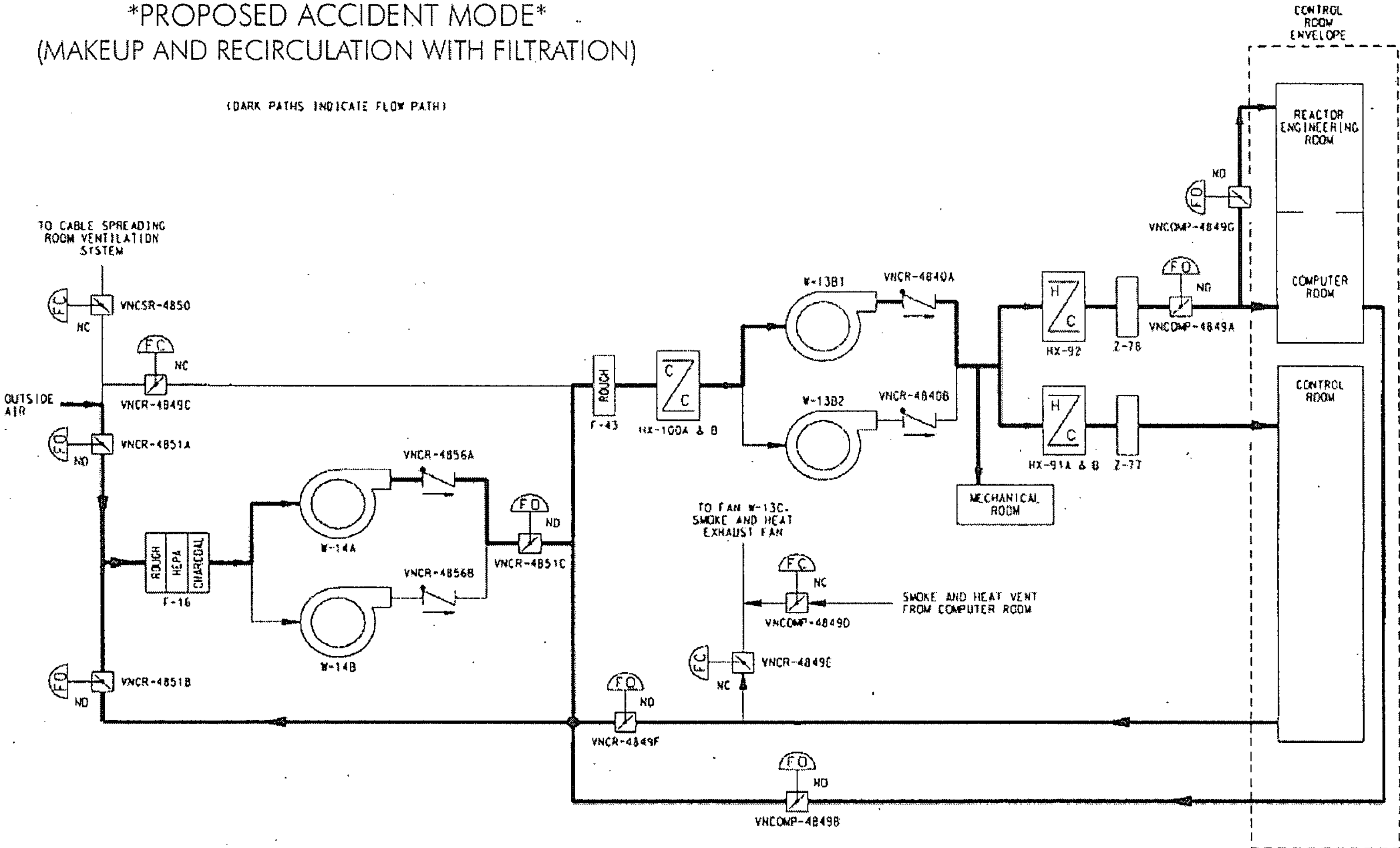
(DARK PATHS INDICATE FLOW PATH)



CONTROL ROOM HVAC

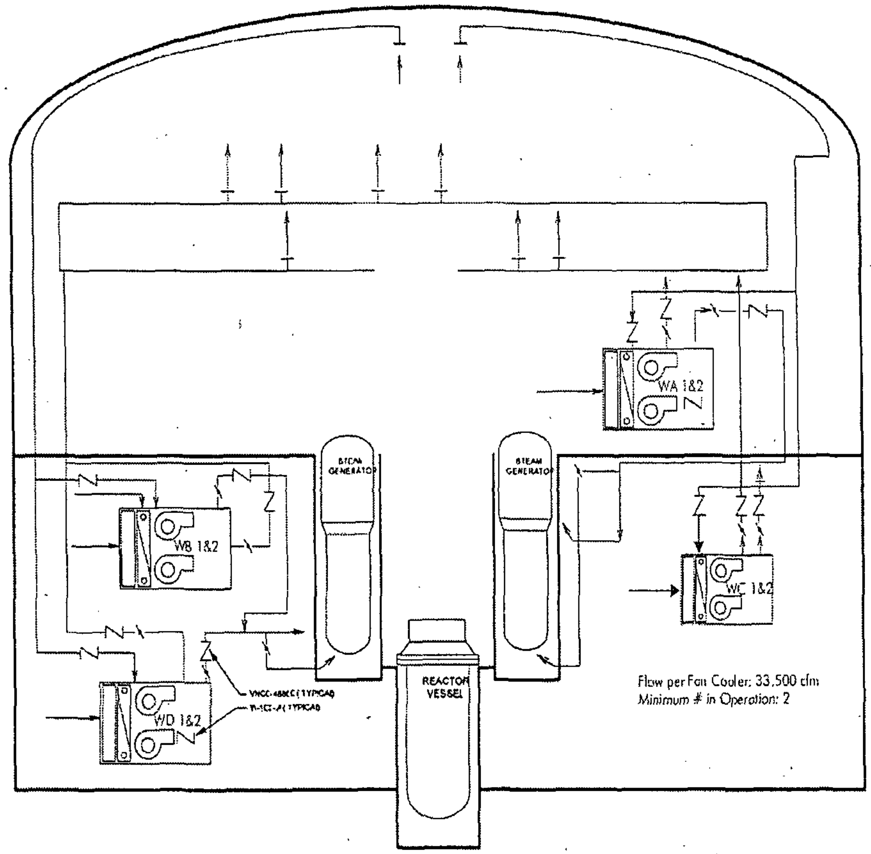
Figure 5

PROPOSED ACCIDENT MODE
(MAKEUP AND RECIRCULATION WITH FILTRATION)



CONTROL ROOM HVAC

Figure 6
Containment Ventilation



ENCLOSURE 3

APPENDIX A

LICENSE AMENDMENT REQUEST 241
TECHNICAL EVALUATION

INPUT ASSUMPTIONS COMPARISON TABLES

POINT BEACH NUCLEAR PLANT

23 Pages Follow

Table A-1 Loss of Coolant Accident (LOCA)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Power Level, MWt	1549	1683	1811	AST models the EPU power level
Iodine chemical form in containment, %				
Elemental	91	4.85	4.85	AST reflects RG 1.183 chemical species vs. RG 1.4
Organic (methyl)	4	0.15	0.15	
Particulate (cesium iodide)	5	95	95	
Containment net free volume, ft^3	1.065E+06	1.0E+06	1.0E+06	AST volume based on updated evaluation
Containment sprayed volume, ft^3	4.75E+5	5.82E+05	5.82E+05	AST volume based on updated evaluation
Fan cooler units				
Number in operation	2	2	2	
Flow rate (per unit), cfm	38,500	33,500	33,500	AST flow extrapolated from measured flow.
Delay time to start, seconds	90	90	90	
Containment leak rate, weight %/day				
0-24 hours	0.4	0.2	0.2	AST leak rate is reduced to minimize offsite and CR doses and eliminate reliance on KI
>24 hours	0.2	0.1	0.1	
Spray operation				
Injection sprays initiated, seconds	90	0	0	No dose impact. No initial activity in the unsprayed volume
Injection sprays terminated, min.	65	60	60	Conservative
Recirculation spray delay, min.	NA	20	20	AST models recirculation spray removal of fission products
Recirculation spray duration, hours	NA	4	3	Recirculating spray is modeled to address the AST fission product release timing.
Spray flow rates, gpm				
Injection	1190	1,111	1,070	AST uses an updated injection spray flow rate
Recirculation	0	900	900	Recirculating spray used for AST
Spray fall height, ft	65.58	65.58	65.58	

Table A-1 Loss of Coolant Accident (LOCA)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Containment Spray removal coefficients, hr^{-1}				
Element iodine				
Injection	20	20	20	
Recirculation	NA	10	9.2	Recirc spray removal of fission products is modeled for AST
Particulate				
Injection	6.02	4.59	4.42	AST reflects reduced spray flow rate
Recirculation	NA	3.72	3.72	Recirculating spray removal of fission products modeled for AST
Containment Spray DF				
Elemental	200	200	200	
Particulate	50	1000	1000	AST methodology requires no DF cut-off of removal. DF of 1000 is conservatively applied to the combined effects of spray and sedimentation.
Sedimentation particulate removal, hr^{-1}	NA	0.1	0.1	AST methodology models sedimentation removal
Containment sump volume, gal	1.97E+05	2.43E+05	2.43E+05	AST volume based on an updated evaluation which reflects plant modifications made, to both units, to minimize hold up in the lower refueling cavity, which increases the sump water volume.
Containment sump pH	≥7.0	≥7.0	≥7.0	
RWST minimum water volume, gal	NA	NA	25,500 gal	AST uses these parameters in the ECCS dose calculation for back-leakage to the RWST. This release pathway was not previously considered.
RWST minimum air volume, ft3	NA	NA	36,100	
RWST minimum temperature, °F	NA	NA	40	
RWST maximum temperature, °F	NA	NA	100	
RWST max boron concentration, ppm	NA	NA	3500 (1)	
Time to initiate RHR recirculation, min	20	0	0.0	AST models ECC system leakage beginning at t=0

(1) Sump pH calculation assumes maximum boron value of 3200 ppm

Table A-1 Loss of Coolant Accident (LOCA)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
ECCS leakage, cc/min PAB (CR/Offsite) RWST	 800/400 NA	 800/800 NA	 300/300 500/500	AST model ECCS back-leakage to the RWST, not modeled in CLB. AST uses a consistent leak rate for both offsite and CR doses.
ECCS leakage iodine airborne % PAB RWST	 10 NA	 10 NA	 10 NA	AST models ECCS back-leakage to the RWST. Activity is released from the RWST in proportion to the air displacement rate, due to diurnal heating and cooling, and the liquid/vapor partition.
Atmospheric Dispersion (χ/Q) Factors, sec/m^3				
Control room, containment surface,				
0 – 2 hours	3.0E-3	1.39E-03	1.39E-03	AST χ/Q values are calculated with ARCON96 and are based on 5 current years of meteorology
2 – 8 hours	3.0E-3	9.80E-04	9.80E-04	
8 – 24 hours	1.9E-3	3.84E-04	3.84E-04	
24 – 96 hours	1.2E-3	3.46E-04	3.46E-04	
96 – 720 hours	4.8E-4	3.02E-04	3.02E-04	
Control room, Auxiliary Building Vent				
0 – 2 hours	1.7E-3	1.80E-03	1.80E-03	AST χ/Q values are calculated with ARCON96 and are based on 5 current years of meteorology
2 – 8 hours	1.7E-3	1.31E-03	1.31E-03	
8 – 24 hours	1.2E-3	5.15E-04	5.15E-04	
24 – 96 hours	6.7E-4	4.03E-04	4.03E-04	
96 – 720 hours	2.3E-4	3.03E-04	3.03E-04	
Control Room, Unit 2 RWST				
0 – 2 hours	NA	NA	9.89E-03	AST models ECCS back-leakage to the RWST. AST χ/Q values are calculated with ARCON96 and are based on 5 current years of meteorology.
2 – 8 hours			7.98E-03	
8 – 24 hours			2.88E-03	
24 – 96 hours			2.75E-03	
96 – 720 hours			2.35E-03	

Table A-2 Steam Generator Tube Rupture (SGTR)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Power Level, MWt	1650	1683	1811	AST models the EPU power level
RCS Iodine Activity (Initial), µCi/gm				
Pre-Accident Spike, DE I-131	50	60	60	AST is consistent with RG 1.183
Accident Initiated Spike, DE I-131	0.8	0.5	0.5	AST is consistent with PBNP proposed Tech Specs
Noble Gas	FSAR Table 14.1.8-4		Table 6 520 µCi/gm DE Xe-133	AST use of DE Xe-133 is consistent with PBNP Tech specs
Alkali Metals	NA	NA	Corresponds to 0.5 µCi/gm DE I-131	AST models the release of alkali metals from the reactor coolant, consistent with RG 1.183
Accident Initiated Iodine Spike Appearance Rate Factor	500	335	335	AST models the rate factor consistent with RG 1.183
Duration of Accident Initiated Spike, hours	1.6	8	8	AST duration is consistent with RG 1.183
Secondary System Activity (Initial), µCi/gm				
Iodine, DE-I-131	1.0	0.1	0.1	AST DE concentration is consistent with PBNP proposed Tech Specs
Alkali metals	NA	NA	Corresponds to 0.1 µCi/gm DE-I-131	AST models the release of alkali metals from the secondary coolant, consistent with RG 1.183
Reactor Coolant Initial Mass, gm	1.1E+08	1.07E+08	1.06E+08	AST uses minimum RCS mass to maximize the nuclide concentrations for the accident initiated spike case.
Steam Generator Initial Mass, gm/SG	3.61 E+07	3.19E+07	2.99E+07	AST uses minimum SG mass to maximize the nuclide concentrations when activity is transferred via the rupture and p/s leakage.
Trip time, sec	NA	148	220	AST models pre and post-trip break flow and steam release
Offsite power	Lost	Lost at time of Rx trip	Lost at time of Rx trip	AST models LOOP concurrent with reactor trip

Table A-2 Steam Generator Tube Rupture (SGTR)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Primary-to-Secondary Leakage Duration for intact SG, hours	8	8	30	AST uses conservative methodology to determine the time for RHR to remove the heat load.
Iodine Species Released to Atmosphere, % Elemental Organic	 100 0	 97 3	 97 3	AST models the iodine chemical species consistent with RG 1.183.
Mass Release Data				
Break flow duration, min	30	30	30	
Ruptured Steam Generator				
Pre-trip break flow, lbm	123,600	26,165	21,300	CLB doesn't model pre-trip break flow. Total AST break flow is comparable to CLB.
Post-trip break flow, lbm		97,435	103,200	
Pre-trip flashed break flow, lbm	NA	5,110	4,690	AST models break flow flashing, consistent with RG 1.183.
Post-trip flashed break flow, lbm	NA	12,520	13,420	
Pre-trip Break Flow Flash Fraction	NA	0.2	0.22	
Post-Trip Break Flow Flash Fraction	NA	0.13)	0.13	
Steam Release Pre-trip Post-trip	 74,000 lbm	 74,000 lbm	 1130 lbm/sec 88,100	The AST model includes a 220 second full power steam release (pre-trip). The post-trip steam release is consistent with the EPU power level.

Table A-2 Steam Generator Tube Rupture (SGTR)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
SG Iodine Partition Factor				
Non-flashed	0.01	0.01	0.01	AST models activity release from non-flashed and flashed rupture flow
Flashed	NA	1.0	1.0	
SG Particulate Retention				
Non-flashed	NA	0.0025	0.0025	AST models the release of alkali metals, and the associated retention factor, for both the flashed and non-flashed break flow
Flashed	NA	1.0	1.0	
Condenser Partition Factor	0.01	0.01	0.01	
Intact Steam Generator				
Primary-to-Secondary Leakage	0.35 gpm	0.35 gpm	1000 gm/min per SG	AST and CLB leak rates are equivalent, based on hot conditions.
Steam Release	NA	NA	1130 lbm/sec (0 – trip)	AST models pre-trip full power steam release
	232,600 lbm (0 - 2 hr)	232,600 lbm (0 – 2 hr)	257,700 lbm (trip – 2 hr)	Consistent with EPU steam release model
	374,300 lbm (2 – 8 hr)	623,800 lbm (2 – 8 hr)	584,000 lbm (2 – 8 hr)	AST models the calculated 2-8 hour steam release. CLB 2 -8 hr release is based on a 2-24 hour steam release, which was linearly interpolated to determine the 2-8 hour release. Releases are not linear with time, but higher earlier.

Table A-2 Steam Generator Tube Rupture (SGTR)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
	NA	NA	866,000 lbm (8 – 24 hr)	AST begins RHR operation at 30 hours, when RHR heat removal equals decay heat, terminating steam release to the environment.
	NA	NA	54,100 lbm/hr (24-30 hr)	
Atmospheric Dispersion (χ/Q) Factors, sec/m^3				
Control room, Unit 2 "A" Safeties				
0 - 2 hours	1.9E-03	4.66E-03	4.66E-03	AST χ/Q values are calculated with ARCON96 and are based on 5 current years of meteorology
2 – 8 hours	1.3E-03	3.40E-03	3.40E-03	
8 - 24 hours	1.3E-03	1.17E-03	1.17E-03	
24 - 96 hours	NA	1.07E-03	1.07E-03	
96 – 720 hours	NA	9.05E-04	9.05E-04	

Table A-3 Locked Rotor (LR)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Power Level, MWt	1650	1683	1811	AST models the EPU power level
Fraction of fuel rods assumed to fail for dose considerations	1.0	1.0	0.3	AST used Westinghouse RAVE methodology to determine rods-in-DNB
Radial Peaking Factor	NA	NA	1.7	AST models the EPU peaking factor
Gap Fractions				
I-131	0.10	0.08	0.08	AST models the gap fractions consistent with RG 1.183, Table 3
Kr-85	0.10	0.10	0.10	
Other Iodines and Noble Gases	0.10	0.05	0.05	
Alkali Metals	NA	0.12	0.12	
Reactor Coolant Activity (Initial)				
Iodine, µCi/gm DE I-131	0.8	0.5	0.5	AST is consistent with the proposed Tech Specs
Noble Gas	1% fuel defect level	1% fuel defect level	520 µCi/gm DE Xe-133	AST use of DE Xe-133 is consistent with PBNP Tech Specs
Alkali Metal	NA	Corresponds to 0.5 µCi/gm DE I-131	Corresponds to 0.5 µCi/gm DE I-131	AST models the release of alkali metals from the reactor coolant, consistent with RG 1.183
Secondary Coolant Activity (Initial)				

Table A-3 Locked Rotor (LR)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Iodine, µCi/gm DE I-131	1.0	0.1	0.1	AST DE concentration is consistent with PBNP proposed Tech Specs
Alkali Metal	NA	Corresponds to 0.1 µCi/gm DE I-131	Corresponds to 0.1 µCi/gm DE I-131	AST models the release of alkali metals from the secondary coolant, consistent with RG 1.183
Primary-to-Secondary Leakage	0.70 gpm total	0.70 gpm total	2000 gm/min total	AST and CLB leak rates are equivalent, based on hot conditions
Steam Release to Environment				
0 – 2 hours, lbm	206,000	204,000	213,295	AST begins RHR operation at 14 hours. RHR heat removal equals decay heat at 30 hours, terminating steam release to the environment.
2 - 8 hours, lbm	434,000	443,000	393,961	
8 -14 hours, lbm	NA	NA	325,084	
14 -30 hours, lbm	NA	NA	561,112	
SG Iodine Partition Factor	0.01	0.01	0.01	
SG Alkali Metal Retention Factor	NA	0.0025	0.0025	AST models the release of alkali metals, and the associated retention factor.
Iodine Species Released to Atmosphere, %				
Elemental	100	97	97	AST iodine chemical species are consistent with RG 1.183
Organic	0	3	3	
RCS Mass, gm	1.1E+08	1.07E+08	1.06E8	AST uses minimum RCS mass to maximize the

Table A-3 Locked Rotor (LR)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
				nuclide concentrations due to failed fuel.
Secondary Side mass, gm total				
0 – 2 hours	7.45E+07	6.38+07	5.98E+07	AST uses minimum SG mass (0-2 hr) to maximize the nuclide concentrations when activity is transferred via p/s leakage. AST models an increase in the SG mass, after 2 hours
> 2 hours		7.37+07	7.37E+07	
Atmospheric Dispersion (χ/Q) Factors, sec/m^3 Control room, Unit 2 "A" Safety Valves				
0 - 2 hours	1.9E-03	4.66E-03	4.66E-03	AST χ/Q values are calculated with ARCON96 and are based on 5 current years of meteorology
2 – 8 hours	1.9E-03	3.40E-03	3.40E-03	
8 - 24 hours	1.3E-03	1.17E-03	1.17E-03	
24 - 96 hours	7.6E-04	1.07E-03	1.07E-03	
96 – 720 hours	2.9E-04	9.05E-04	9.05E-04	

Table A-4 Main Steam Line Break (MSLB)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Power Level, MWt	1650	1683	1811	AST models the EPU power level
Reactor Coolant Activity (Initial)				
Pre-Accident Iodine Spike, μCi/gm DE I-131	50	60	60	AST is consistent with RG 1.183
Accident-Initiated Iodine Spike, μCi/gm DE I-131	0.8	0.5	0.5	AST is consistent with PBNP proposed Tech Specs
Noble Gas	1.0% fuel defect level	1.0% fuel defect level	520 μCi/gm DE Xe-133	AST use of DE Xe-133 is consistent with PBNP Tech Specs
Alkali Metal	NA	Corresponds to 0.5 μCi/gm DE I-131	Corresponds to 0.5 μCi/gm DE I-131	AST models the release of alkali metals from the reactor coolant, consistent with RG 1.183
RCS Accident-Initiated Iodine Appearance Rate Spike Factor	500	500	500	AST models the rate factor consistent with RG 1.183
Duration of Accident-Initiated Iodine Spike, hours	1.6	4.0	4	AST models the spike duration consistent with the available gap inventory.
RCS Mass, gm	1.10E8	1.07E+08	1.06E+08	AST uses minimum RCS mass to maximize the nuclide concentrations for the accident initiated spike case
Secondary Coolant Activity (Initial)				
Iodine, μCi/gm DE I-131	1.0	0.1	0.1	AST concentration is consistent with PBNP proposed Tech Specs

Table A-4 Main Steam Line Break (MSLB)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Alkali Metal	NA	Corresponding to 0.1 μCi/gm DE I-131	Corresponding to 0.1 μCi/gm DE I-131	AST models the release of alkali metals from the secondary coolant, consistent with RG 1.183
Primary-to-Secondary Leakage, per SG	0.35 gpm	0.35 gpm	1000 gm/min	AST and CLB leak rates are equivalent, based on hot conditions
Faulted SG				
Time to Release Initial SG water Mass, min	15	2	2	AST models a conservatively short (for CR doses) time to boil dry the Faulted SG
Secondary Side water mass, gm/sg	3.61E+07	5.7E+07	5.7E+07	AST models conservatively high zero power mass to maximize the initial SG nuclide inventory
Steam Release Initial SG mass Primary-to-secondary leakage (after initial release, until cold conditions).	3.61E+07 (0-15 min) 0.35 gpm	5.7E+07 (0-2 min) 0.35 gpm	5.7E+07 (0-2 min) 1000 gm/min	AST models a conservatively high zero power mass to maximize the initial SG nuclide inventory
Time to Cool RCS Below 212EF (and stop releases from Faulted SG), hours	8 hours	30 hours	60 hours	AST models a conservatively long time to cool the RCS below 212EF. CLB modeled time to begin RHR operation.
SG Iodine Partition Factor	1.0	1.0	1.0	
Iodine Species Released to Atmosphere, %				

Table A-4 Main Steam Line Break (MSLB)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Elemental	100	97	97	AST models the iodine chemical species, released from the secondary system, consistent with RG 1.183
Organic	0	3	3	
Intact SG				
Steam Releases, lbm				
0 – 2 hours	212,000	213,000	221,153	
2 – 8 hours	405,000	413,000	1,048,064	AST models a conservatively long time to cool the RCS below 212EF.
2 – 24 hours	NA	NA		
24 - 30 hours	NA	NA	201,570	
SG Iodine Partition Factor	0.01	0.01	0.01	
SG Particulate Retention Factor	NA	0.0025	0.0025	AST models the release of alkali metals, consistent with RG 1.183
Intact SG mass, gm	3.61E+07	3.19E+07	2.99E+07	AST uses minimum SG mass to maximize the nuclide concentrations when activity is transferred via p/s leakage
Atmospheric Dispersion (χ/Q) Factors, sec/m^3				

Table A-4 Main Steam Line Break (MSLB)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Control room, Unit 2 Containment				CLB modeled activity releases from the containment surface
0 - 2 hours	2.1E-03	NA	NA	
2 – 8 hours	2.1E-03	NA	NA	
8 - 24 hours	1.3E-03	NA	NA	
24 - 96 hours	NA	NA	NA	
96 – 720 hours	NA	NA	NA	
Control room, Unit 2 "A" Safeties				
0 - 2 hours	NA	4.66E-03	4.66E-03	AST modeled the intact SG activity releases from the Unit 2 safety valves. AST X/Q values are calculated with ARCON96 and are based on 5 current years of meteorology
2 – 8 hours	NA	3.40E-03	3.40E-03	
8 - 24 hours	NA	1.17E-03	1.17E-03	
24 - 96 hours	NA	1.07E-03	1.07E-03	
96 – 720 hours	NA	9.05E-04	9.05E-04	
Control room, Unit 2 Containment Facade Penetration				
0 - 2 hours	NA	1.87E-02	1.87E-02	AST modeled the faulted SG activity releases from the Unit 2 steam line penetration in the containment facade. AST X/Q values are calculated with ARCON96 and are based on 5 current years of meteorology
2 – 8 hours	NA	1.50E-02	1.50E-02	
8 - 24 hours	NA	5.11E-03	5.11E-03	
24 - 96 hours	NA	4.94E-03	4.94E-03	

Table A-4 Main Steam Line Break (MSLB)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
96 – 720 hours	NA	4.23E-03	4.23E-03	

Table A-5 *Control Rod Ejection (CRDE)*				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Power Level, MWT	1650	1683	1811	AST models the EPU power level
Failed Fuel Fraction	0.10	0.10	0.10	
Gap Fractions				
Iodine	0.10	0.10	0.10	AST models the gap fractions consistent with RG 1.183, Appendix H.
Noble Gas	0.10	0.10	0.10	
Alkali Metals	NA	0.12	0.12	
Melted Fuel Fraction	0.0025	0.0025	0.0025	
Radial Peaking Factor	1.8	1.8	1.7	AST models the EPU peaking factor
Fraction of Activity Available from Melted Fuel				
Via Containment Leakage				
Iodine	0.25	0.25	0.50	AST does not model instantaneous iodine plate-out, as recommended by RG 1.183, adding conservatism
Noble Gas	1.0	1.0	1.0	
Alkali Metals	NA	0.25	0.50	AST models the release of alkali metals, consistent with RG 1.183
Via Primary–to-Secondary Leakage				
Iodine	0.50	0.50	0.50	

Table A-5 Control Rod Ejection (CRDE)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Noble Gas	1.0	1.0	1.0	
Alkali Metals	NA	0.50	0.50	AST models the release of alkali metals, consistent with RG 1.183
Reactor Coolant Activity (Initial)				
Iodine, µCi /gm D.E. I-131	0.8	0.5	0.5	AST models reduced activity, consistent with the proposed Tech Spec
Noble Gas	1.0% fuel defect level	1.0% fuel defect level	520 µCi/gm DE Xe-133	AST use of DE Xe-133 is consistent with the PBNP Tech Spec
Alkali Metals	NA	1.0% fuel defect level	Corresponds to 0.5 µCi/gm DE I-131	AST models the release of alkali metals from the reactor coolant, consistent with RG 1.183
Secondary Coolant Activity (Initial)				
Iodine, µCi/gm DE I-131	1.0	0.1	0.1	AST DE concentration is consistent with PBNP proposed Tech Specs
Alkali Metal	NA	20% of primary concentration	Corresponding to 0.1 µCi/gm DE I-131	AST models the release of alkali metals from the secondary coolant, consistent with RG 1.183
Containment Net Free Volume, ft^3	1.065E+06	1.0E+06	1.00E+06	AST volume based on updated evaluation
Containment leak rate, weight %/day				
0 – 24 hours	0.4	0.2	0.2	AST leak rate is consistent with the LBLOCA.
> 24 hours	0.2	0.1	0.1	

Table A-5
Control Rod Ejection (CRDE)

Parameter	CLB	2007 Submittal	AST Submittal	Remarks
Iodine chemical form in containment, %				
Elemental	100	4.85	4.85	AST models the iodine chemical form in containment consistent with RG 1.183
Organic	NA	0.15	0.15	
Particulate (cesium iodide)	NA	95	95	
Spray removal in containment	Not credited	Not credited	Not credited	
Sedimentation removal in containment				
Iodines	NA	NA	NA	
Alkali metals	NA	0.1 hr^{-1}	0.1hr^{-1}	AST models sedimentation removal of particulates, consistent with the LBLOCA.
Primary-to-Secondary Leakage, Total	0.70 gpm	0.70 gpm	2000 gm/min	AST and CLB leak rates are equivalent, based on hot conditions
Steam Releases to Environment , ibm				
0 – 2 hours	158,200	204,000	213,295	AST models steam releases beyond 8 hours, until RHR can take over decay heat removal
2 – 8 hours	428,000	443,000	719,045	
8 – 14 hours	NA	NA		
14 - 30 hours			561,112	
SG Iodine Partition Factor	0.01	0.01	0.01	
SG Alkali Metal Retention Factor	NA	0.0025	0.0025	AST models the release of alkali metals from the secondary side, consistent with RG 1.183

Table A-5 Control Rod Ejection (CRDE)				
Parameter	**CLB**	***2007 Submittal***	**AST Submittal**	**Remarks**
Iodine Chemical form released to Atmosphere, %				
Elemental	100	97	97	AST models the iodine chemical species released from the secondary system, consistent with RG 1.183
Organic	NA	3	3	
RCS Mass, gm	1.1E+08	1.07E+08	1.06E+-8	AST uses minimum RCS mass to maximize the nuclide concentrations due to failed and melted fuel.
Total SG Mass, gm	7.18E+07	6.38E+07	5.98E+07	AST use minimum SG mass to maximize the nuclide concentrations when activity is transferred via p/s leakage

Table A-5 Control Rod Ejection (CRDE)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Atmospheric Dispersion (χ/Q) Factors, sec/m^3				
Control room, Unit 2 Containment Surface				
0 - 2 hours	2.1E-03	1.39E-03	1.39E-03	AST X/Q values are calculated with ARCON96 and are based on 5 current years of meteorology.
2 – 8 hours	2.1E-03	9.80E-04	9.80E-03	
8 - 24 hours	1.3E-03	3.84E-04	3.84E-03	
24 - 96 hours	NA	3.46E-04	3.46E-03	
96 – 720 hours	NA	3.02E-04	3.02E-04	
Control room, Unit 2 "A" Safeties				
0 - 2 hours	NA	4.66E-03	4.66E-03	AST X/Q values are calculated with ARCON96 and are based on 5 current years of meteorology.
2 – 8 hours	NA	3.40E-03	3.40E-03	
8 - 24 hours	NA	1.17E-03	1.17E-03	
24 - 96 hours	NA	1.07E-03	1.07E-03	
96 – 720 hours	NA	9.05E-04	9.05E-04	

Table A-6 Fuel Handling Accident (FHA)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Power Level, MWt	1650	1683	1811	AST models the EPU power level
Fuel Damage, # of assemblies	1	1	1	
Radial Peaking Factor	1.8	1.8	1.7	AST models the EPU peaking factor
Time from Shutdown before fuel movement	65 hours	65 hours	65 hours	
Iodine chemical form in pool				
Elemental	99.85%	99.85%	99.85%	
Organic (methyl)	0.15%	0.15%	0.15%	
Gap Fractions				
I-131	0.08	0.12	0.12	AST assumes that the damaged assembly exceeds the criteria of RG 1.183, footnote 11 and models the FHA iodine and noble gap fractions consistent with the recommendation of NUREG/CR-5009.
Iodine	0.05	0.10	0.10	
Kr-85	0.10	0.30	0.30	
Noble Gas	0.05	0.10	0.10	
Water depth	23 feet	23 feet	23 feet	
Overall Pool Iodine Scrubbing Factor	200	200	200	
Filter Efficiency	No filtration assumed	No filtration assumed	No filtration assumed	
Isolation of Release	No isolation assumed	No isolation assumed	No isolation assumed	

Table A-6 Fuel Handling Accident (FHA)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Duration of activity release, hours	2	2	2	
Delay to Switch CR HVAC from Normal to Accident Mode`, min	10	10	10	
Atmospheric Dispersion (χ/Q) Factors, sec/m^3				
Control room, Unit 2 Purge Stack				
0 - 2 hours	5.76E-3	6.94E-03	6.94E-03	AST X/Q values are calculated with ARCON96 and are based on 5 current years of meteorology
2 – 8 hours	NA	NA	NA	
8 - 24 hours	NA	NA	NA	
24 - 96 hours	NA	NA	NA	
96 – 720 hours	NA	NA	NA	

Table A-7 Reactor Vessel Head Drop (RVHD)				
Parameter	**CLB**	**2007 Submittal**	**AST Submittal**	**Remarks**
Fuel Damaged, %	100	100	100	
Fuel Melt, %	0	0	0	
Time from Shutdown to Movement	100 hours	Immediate	Immediate	
Iodine Gap Fraction	0.08	0.05	0.05	AST gap fraction consistent with RG 1.183, Table 2
Iodine Form Released to Atmosphere	Elemental	Elemental	Elemental	
Recirculation Initiation Time	Immediate	Immediate	Immediate	
ECCS leakage, cc/min PAB (CR/Offsite) RWST	 800/400 NA	 800/800 NA	 300/300 500	AST model ECCS back-leakage to the RWST, not modeled in CLB. AST uses a consistent leak rate for both offsite and CR doses.
ECCS leakage iodine airborne, % PAB RWST	 10 NA	 10 NA	 10 NA	AST models ECCS back-leakage to the RWST. Activity is released from the RWST in proportion to the air displacement rate, due to diurnal heating and cooling, and the liquid/vapor partition.
Containment Sump Volume, gal	2.43E+05	2.43E+05	2.43E+05	
RWST Minimum Water Volume, gal	NA	NA	25,500	AST uses these parameters in the dose calculation for RHR system back-leakage to the RWST. This release pathway was not previously considered.
RWST Minimum Air Volume, gal	NA	NA	270,000	
RWST Minimum Temperature, °F	NA	NA	40	
RWST Maximum Temperature, °F	NA	NA	100	
Boron Concentration for ECCS RWST	NA	NA	3500 ppm	

ENCLOSURE 4

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

POINT BEACH NUCLEAR PLANT

8 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.	Enclosure 1 provides a summary of the changes, including the justification for each change. Enclosure 3 provides the AST Technical Evaluation, including specific details on the individual radiological analyses in Section 6. ARCON96 meteorological data files and RADTRAD input decks are being submitted on CDs under separate cover letter.
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.	This item is applicable only to BWRs. PBNP is a PWR, therefore this item is not applicable to PBNP.

RIS 2006-04 ISSUE	ADDRESSED BY
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 part of the ventilation filter testing program in Section 5 of the TS.	The Control Room Ventilation System (VNCR) is credited in the control room dose analyses. The VNCR system is classified as non-safety related, however selected components such as the control room charcoal filter fans, the control room recirculation fans, damper controllers and the charcoal/HEPA/roughing filter have been upgraded to augmented quality status. For the accident mode of operation, the two radiation monitor actuation signals are diverse and the radiation monitors are augmented quality status. The containment isolation actuation signals are safety-related and redundant. The fans, dampers and filter are currently tested in accordance with Surveillance Requirements (SR) 3.7.9.1 through SR 3.7.9.6 of the Point Beach Technical Specifications (see page 3.7.9-2 of Enclosure 2). FPL Energy Point Beach will modify the VNCR system to create a new alignment for the accident mode that provides a combination of filtered outside air and filtered recirculation air. In order to address the requirements of Section 5.1.2 of Regulatory Guide 1.183, the VNCR system will be modified to ensure redundancy of active components and to auto-start the system from the safety related diesel generators on loss of offsite power for the new system alignment. The system is currently credited in the CLB and has a backup emergency diesel generator (EDG) power supply that may be manually aligned. Additional details regarding the VNCR system and proposed modifications are provided in Enclosure 3 of this LAR.

RIS 2006-04 ISSUE	ADDRESSED BY
	The Primary Auxiliary Building ventilation (VNPAB) system is non-safety related, but credited in the control room dose analysis for LOCA. VNPAB fans are powered by safety-related power supplies with diesel generator backup. New control room operator actions are required to restore the VNPAB within 30 minutes following the alignment of RHR to containment sump recirculation mode of operation. If a LOCA occurs coincident with a LOOP, the VNPAB system will be manually restarted to ensure that the PAB vent stack is the source of the release associated with the ECCS leakage phase of the event. This configuration is an input assumption for the LOCA radiological analysis. FPL Energy Point Beach will modify the VNPAB to provide redundancy and establish augmented quality for all active components required to implement AST. The LOCA radiological analysis does not credit VNPAB filtration, so no additional Technical Specification filter testing is proposed.
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 stated that the unfiltered inleakage assumption in the control room habitability analyses was non-conservative. The 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 (VNCR accident mode operation) will result in unfiltered inleakage value less than the limiting 200 cfm unfiltered inleakage assumption associated with the LOCA and MSLB radiological analyses. This will allow control room habitability systems to be configured, operated and maintained in accordance with the revised facility design and licensing bases.

RIS 2006-04 ISSUE	ADDRESSED BY
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 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:	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 4.0 of the AST Technical Evaluation, Enclosure 3. The atmospheric dispersion (χ/Q) values for the PBNP EAB and the LPZ are those from the CLB. These values were developed from the guidance provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and meteorological data collected at the site's primary tower from January 1,1991, through December 31, 1993. FPL Energy Point Beach has assessed these χ/Q values against χ/Q values generated using the PAVAN computer code with the meteorological data collected at PBNP from September 2000 to September 2005 and determined the CLB χ/Q's are conservative.
• 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 4.0.
• 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 being provided under a separate cover letter. Assumptions and atmospheric relative concentrations are included in Enclosure 3, Section 4.0. The electronic data has been verified to be properly converted and formatted.

RIS 2006-04 ISSUE	ADDRESSED BY
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 total ECCS recirculation leakage modeled in the analysis is 0.21 gpm (i.e., the current analysis value of 400 cc/min total ECCS leakage outside the containment is doubled to 800 cc/min). Of the 800 cc/min total ECCS recirculation leakage, 300 cc/min is assumed to leak into the PAB, and 500 cc/min is assumed to leak back to the RWST. The historical data for ECCS leakage, collected from the PBNP Leakage Reduction and Preventive Maintenance Program, are less than the current analysis limit of 400 cc/min. The assumed amount of iodine that may become airborne from ESF leakage to the PAB is 10%. For ECCS leakage to RWST, see Enclosure 3, Section 6.1.
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 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.	Consistent with current licensing basis, the radiological analysis for RVHD accident assumes containment closure conditions and the FHA analysis assumes the containment is open.

RIS 2006-04 ISSUE	ADDRESSED BY
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 leak rate used in the accident analysis has significant margin to the operational leak rate limit, when compared on a consistent density basis. The accident-induced primary-to-secondary leak rate, modeled in the LR, MSLB, SGTR and CRDE dose analyses, is significantly greater than the operational leak rate. The assumed accident-induced leak rate is 0.7 gpm, total (TS 5.5.8, "Steam Generator (SG) Program," Accident induced leakage performance criterion of 500 gpd per SG). The assumed density, for conversion to a mass leak rate, is 47 lbm/ft^3 (full power operation), and the corresponding leak rate is 1000 gm/min per SG.
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 decontamination factor (DF) for the FHA 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. See Enclosure 3, Section 6.4 and Section 6.5.

RIS 2006-04 ISSUE	ADDRESSED BY
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 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.	The accidents use the "thyroid dose" conversion factors consistent with the TS definition.

RIS 2006-04 ISSUE	ADDRESSED BY
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.	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 ft3 with a sprayed fraction of 58.2 percent of the total (5.82E5 ft3).

ENCLOSURE 6

FPL ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

WCAP-16259 SAFETY EVALUATION REPORT COMPLIANCE
(NON-PROPRIETARY)

DATED SEPTEMBER, 2008

9 pages follow

Westinghouse Non-Proprietary Class 3

WEP-08-104 NP–Attachment

WCAP-16259 Safety Evaluation Report Compliance

September, 2008

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

WCAP-16259 Safety Evaluation Report Compliance

A RETRAN-SPNOVA-VIPRE (RAVE)

The NRC Safety Evaluation Report (SER) can be found in the front of WCAP-16259-P-A (Reference A.5). This SER stipulates a number of conditions and limitations on the use for licensing basis calculations. The following is a review of these SER restrictions and requirements.

Table A-1: RETRAN-SPNOVA-VIPRE (RAVE)

Limitations, Restrictions and Conditions
1. Consistent with the guidance contained in Generic Letter 88-16, "Removal of Cycle-Specific Parameters Limits from Technical Specifications," a methodology that is used in the evaluation of the cycle-specific safety limits and plant safety analyses needs to be incorporated into the technical specification (TS) list of references. Therefore, the implementation of RAVE on a plant-specific basis required a TS amendment by the plant when the RAVE methodology is first implemented for that plant.
Compliance Compliance was demonstrated by adding WCAP-16259-P-A (Reference A-5) to the PBNP Technical Specifications change package.
2. Because of competing effects between the coupled computer codes, the most conservative assumptions will, in many cases, no longer be obvious. Sensitivity studies will need to be performed to determine the most conservative plant conditions. Since different core designs may exhibit different sensitivities, the first implementation of the RAVE sensitivity studies should be performed to ensure that the limiting conditions have been identified. The sensitivity results will accompany the analyses using the RAVE methodology whenever the RAVE methodology is first implemented for a plant and must be presented to the NRC staff for review and approval.
Compliance Compliance was demonstrated by re-performing the sensitivity cases set forth in WCAP-16259-P-A for the Locked Rotor event for PBNP and the results which contain information considered proprietary to Westinghouse are included.
3. As support for the TS amendment, licensees implementing RAVE should provide justification that SPNOVA, VIPRE and RETRAN computer codes and methodology are approved for use in compliance with the conditions identified in the NRC staff SEs. The methodology for use of the VIPRE code shall be considered to be reviewed and approved for use in the RAVE methodology if all three applications of VIPRE have been reviewed and approved by the NRC staff. The three applications of VIPRE are the whole-core model, the DNBR model, and the post-CHF fuel heat-up model. **If a specific plant has not been licensed for the use of the computer codes and methodology that are utilized by RAVE, then the licensee will need to take appropriate licensing action for application of these computer codes. Licensees will need to verify that the conditions and limitations imposed on each of the three NRC approved codes (SPNOVA, RETRAN, and VIPRE), encompassing the RAVE methodology, will continue to be satisfied each time the RAVE methodology is used.**

Compliance SPNOVA (References A-1 and A-2), VIPRE (Reference A-3) and RETRAN (Reference A-4) have already been generically approved for use at Westinghouse plants, and are applicable to PBNP. Topical report references for the SPNOVA, RETRAN and VIPRE models are included as part of this licensing package (see Enclosure 7). All three applications of VIPRE were used consistently with the VIPRE models described in WCAP-16259-P-A and WCAP-14565-P-A, which were reviewed and approved by the NRC.
4. Westinghouse submitted analyses showing that for post-CHF core heat-up, VIPRE input, as modified by Westinghouse and FACTRAN, produce virtually identical results. Therefore, the NRC staff considers VIPRE to be equivalent to FACTRAN for performing post-CHF core heat-up calculations. As is permitted for FACTRAN, VIPRE can be used to show compliance with acceptance criteria for peak cladding temperature for a locked rotor event, fuel melting, and pellet enthalpy criteria as well as for DNBR evaluation. Neither VIPRE nor FACTRAN include the time-dependent physical changes that may occur in a fuel rod at elevated temperatures. Therefore, VIPRE cannot be used to predict such failures and another fuel code should be used to predict mechanical behavior.
Compliance The VIPRE post-CHF model was used only for performing post-CHF core heat-up (PCT) calculation to show compliance with acceptance criteria for peak cladding temperature for a locked rotor event and was not used to predict the mechanical behavior.
5. The code option selected for use with whole-core VIPRE model may not be conservative for calculation of reactivity feedback for elevated steam void fractions. Westinghouse performed sensitivity studies which demonstrated that the reactor power calculated by the RAVE methodology is insensitive to assumptions for core voiding up to a maximum steam void fraction of 30 percent. If the maximum void fraction in any RAVE reactivity feedback calculation exceeds 30 percent, additional justification will need to be provided for the steam/water separation model utilized in the VIPRE whole-core model to the staff for additional review of that application of RAVE.
Compliance For the events that require a separate hot rod calculation (DNBR and PCT), the VIPRE core feedback calculations are performed with the core conditions satisfying the 30% void fraction limit identified in the RAVE (WCAP-16259-P-A) SER. This is done to predict a conservative nuclear power response during the transient. The 30% void fraction limit was exceeded for the locked rotor peak pressure event due to the use of conservative assumptions that maximize pressure response. However, the impact of exceeding this void fraction limit was investigated and it was determined to be conservative with respect to overpressurization. The pressure penalty associated with the increased voiding in the core (which in turn increases the pressurizer surge) is greater than the benefit (reduction) seen in the nuclear power response due to higher voiding in the core. Therefore, exceeding the 30% void limit is conservative for the locked rotor peak pressure event and provides more limiting peak pressure results compared to the case which does not exceed the 30% void fraction limit.

References

A-1 Chao, Y. A., et al., *SPNOVA – A Multidimensional Static and Transient Computer Program for PWR Core Analysis*, WCAP-12394-A (Proprietary) and WCAP-12983-A (Nonproprietary), June 1991.

A-2 Letter from Liparulo, N.J. (Westinghouse) to Jones, R. C., (NRC), *Process Improvement to the Westinghouse Neutronics Code System*, NTD-NRC-96-4679, March 29, 1996.

A-3 WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Non-Proprietary), *VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis*, Sung, Y. X., et al., October 1999.

A-4 WCAP-14882-P-A, *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses*, D. S. Huegel, et al., April 1999.

A-5 WCAP-16259-P-A (Proprietary) and WCAP-16259-A (Non-Proprietary), *Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis*, Beard, C. L., et al., August 2006.

WCAP-16259 Safety Evaluation Report Compliance

Westinghouse compliance with the NRC SER conditions for WCAP-16259 is addressed in Appendix A. To show compliance with SER item 2, PBNP specific sensitivity studies have been performed for the Locked Rotor event.

2. **Because of competing effects between the coupled computer codes, the most conservative assumptions will, in many cases, no longer be obvious. Sensitivity studies will need to be performed to determine the most conservative plant conditions. Since different core designs may exhibit different sensitivities, the first implementation of the RAVE sensitivity studies should be performed to ensure that the limiting conditions have been identified. The sensitivity results will accompany the analyses using the RAVE methodology whenever the RAVE methodology is first implemented for a plant and must be presented to the NRC staff for review and approval.**
Compliance Compliance was demonstrated by re-performing the sensitivity cases set forth in WCAP-16259-P-A for the Locked Rotor event for PBNP and the results are included **below** for the staff to review.

1.0 Locked Rotor Sensitivity Studies

1.1 Locked Rotor Rods-In-DNB

Sensitivity studies were performed to determine the conservative direction of the key analysis inputs (Reference 1). The reference limiting case was established based on the results of the sensitivity cases. Table A-1 presents the sensitivity cases that were performed, consistent with Reference 1, to establish the reference limiting locked rotor rods-in-DNB case. Results of the sensitivity cases are discussed below. [

]^{a,c}

1. [

]^{a,c}

2. [

]^{a,c}

3. [

] a,c

4. [

] a,c

5. [

] a,c

6. [

] a,c

7. [

] a,c

8. [

a,c

9. [

] a,c

10. [

] a,c

1.2 Locked Rotor – Peak Pressure/Peak Clad Temperature

Sensitivity studies were performed to determine the conservative direction of the key analysis inputs (Reference 1). The reference limiting case was established based on the results of the sensitivity cases. Table A-2 presents the sensitivity cases that were performed, consistent with Reference 1, to establish the reference limiting locked rotor peak pressure case. Results of the sensitivity cases are discussed below.

1. [

   a,c

2. [

   ] a,c

3. [

   ] a,c

4. [

   ] a,c

5. [

   ] a,c

References

1. WCAP-16259-P-A (Proprietary) and WCAP-16259-A (Non-Proprietary), *Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis*, Beard, C. L., et al., August 2006

A-1: Results of Sensitivity Study for Locked Rotor Rods-in-DNB Event

a, c

Table A-2: Results of Sensitivity Study for Locked Rotor Peak RCS Pressure Event

a, c

ENCLOSURE 7

FPL ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

SAFETY EVALUATION COMPLIANCE REPORTS FOR WCAP-10965-P-A (SPNOVA), WCAP-14565-P-A (VIPRE), AND WCAP-14882-P-A (RETRAN)

5 Pages Follow

Enclosure 7

Safety Evaluation Report Compliance for RETRAN, ANC (SPNOVA) and VIPRE

This Enclosure is a summary of NRC-approved codes and methods. The enclosure addresses compliance with the limitations, restrictions, and conditions specified in the approving safety evaluation of the applicable codes and methods.

Safety Evaluation Report Compliance Summary			
Subject	**Topical Report (Reference) / Date of NRC Acceptance**	**Code(s)**	**Limitation, Restriction, Condition**
Non-LOCA Safety Analysis	WCAP-14882-P-A (Reference 1) / February 11, 1999	RETRAN	Yes
Multi-dimensional Neutronics	WCAP-10965-P-A (Reference 2) / June 23, 1986	ANC (SPNOVA)	None for Non-LOCA Transient Analysis
Non-LOCA Thermal / Hydraulics	WCAP-14565-P-A (Reference 3) / January 19, 1999	VIPRE	Yes

References

1. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," D. S. Huegel, et al., April 1999.

2. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," Y. S. Liu, et al., September 1986.

3. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, et al., October 1999.

1. RETRAN for Non-LOCA Safety Analysis

Limitations, Restrictions, and Conditions

1. ***"The transients and accidents that Westinghouse proposes to analyze with RETRAN are listed in this SER (Table 1) and the NRC staff review of RETRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification."***

Justification

The transients listed in Table 1 of the SER are:

- Feedwater system malfunctions
- Excessive increase in steam flow
- Inadvertent opening of a steam generator relief or safety valve
- Steam line break
- Loss of external load/turbine trip
- Loss of offsite power
- Loss of normal feedwater flow
- Feedwater line rupture
- Loss of forced reactor coolant flow
- Locked reactor coolant pump rotor/sheared shaft
- Control rod cluster withdrawal at power
- Dropped control rod cluster/dropped control bank
- Inadvertent increase in coolant inventory
- Inadvertent opening of a pressurizer relief or safety valve
- Steam generator tube rupture

The transients analyzed for PBNP using RETRAN are:

- Excessive increase in steam flow (PBNP UFSAR Section 14.1.7)
- Steam line break (PBNP UFSAR Section 14.2.5)
- Loss of external electrical load (PBNP UFSAR Section 14.1.9)
- Loss of all alternating current power to the station auxiliaries (PBNP UFSAR Section 14.1.11)
- Loss of normal feedwater flow (PBNP UFSAR Section 14.1.10)
- Loss of reactor coolant flow (PBNP UFSAR Section 14.1.8)
- Locked rotor accident (PBNP UFSAR Section 14.1.8)
- Uncontrolled rod withdrawal at power (PBNP UFSAR Section 14.1.2)

As each transient analyzed for PBNP using RETRAN matches one of the transients listed in Table 1 of the SER, additional justification is not required.

Limitations, Restrictions, and Conditions
2. ***"WCAP-14882 describes modeling of Westinghouse designed 4-, 3, and 2-loop plants of the type that are currently operating. Use of the code to analyze other designs, including the Westinghouse AP600, will require additional justification."***   <u>Justification</u>   The PBNP consists of a two 2-loop Westinghouse-designed units that were "currently operating" at the time the SER was written (February 11, 1999). Therefore, additional justification is not required.   **3.** ***"Conservative safety analyses using RETRAN are dependent on the selection of conservative input. Acceptable methodology for developing plant-specific input is discussed in WCAP-14882 and in WCAP-9272-P-A. Licensing applications using RETRAN should include the source of and justification for the input data used in the analysis."***   <u>Justification</u>   The input data used in the RETRAN analyses performed by Westinghouse came from both PBNP and Westinghouse sources. Assurance that the RETRAN input data is conservative for PBNP is provided via Westinghouse's use of transient-specific analysis guidance documents. Each analysis guidance document provides a description of the subject transient, a discussion of the plant protection systems that are expected to function, a list of the applicable event acceptance criteria, a list of the analysis input assumptions (e.g., directions of conservatism for initial condition values), a detailed description of the transient model development method, and a discussion of the expected transient analysis results. Based on the analysis guidance documents, conservative plant-specific input values were requested and collected from the responsible PBNP and Westinghouse sources. Consistent with the Westinghouse Reload Evaluation Methodology described in WCAP-9272-P-A (Reference 1), the safety analysis input values used in the PBNP analyses were selected to conservatively bound the values expected in subsequent operating cycles.

Reference

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," S. L. Davidson (Ed.), July 1985.

2. VIPRE for Non-LOCA Thermal/Hydraulics

Limitations, Restrictions, and Conditions
1. ***"Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal."*** Justification The WRB-1 correlation with a 95/95 correlation limit of 1.17 was used in the DNB analyses for the PBNP 14×14 422V+ fuel. The use of the WRB-1 DNB correlation is based on the notification change which introduces the 14x14 422V+ mid-grid design (NPL 97-0538, CAW-97-1166). The basic change is reverting back to the larger OD fuel rod as in standard fuel but with a new Low Pressure Drop mid-grid design. The applicability of WRB-1 to the LPD mid-grid was justified under FCEP (WCAP-12488-A). The use of the plant specific hot channel factors and other fuel dependent parameters in the DNB analysis for the PBNP 422V+ fuel were justified using the same methodologies as for previously approved safety evaluations of other Westinghouse two-loop plants using the same fuel design.
2. ***"Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE."*** Justification The core boundary conditions for the VIPRE calculations for the 422V+ fuel are all generated from NRC-approved codes and analysis methodologies. Conservative reactor core boundary conditions were justified for use as input to VIPRE. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in WCAP-9272-P-A (Reference 1).
3. ***"The NRC Staff's generic SER for VIPRE set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification."*** Justification As discussed in response to Condition 1, the WRB-1 correlation with a limit of 1.17 was used in the DNB analyses of 422V+ fuel for PBNP. For conditions where WRB-1 is not applicable, the W-3 DNB correlation was used with a limit of 1.30 (1.45, for pressures between 500 psia and 1,000 psia).

Limitations, Restrictions, and Conditions
4. ***"Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC Staff's generic review of VIPRE did not extend to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC Staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained."***
Justification For application to PBNP safety analysis, the usage of VIPRE in the post-critical heat flux region is limited to the peak clad temperature calculation for the locked rotor transient. The calculation demonstrated that the peak clad temperature in the reactor core is well below the allowable limit to prevent clad embrittlement. VIPRE modeling of the fuel rod is consistent with the model described in WCAP-14565-P-A and included the following conservative assumptions: • DNB was assumed to occur at the beginning of the transient, • Film boiling was calculated using the Bishop-Sandberg-Tong correlation, • The Baker-Just correlation accounted for heat generation in fuel cladding due to zirconium-water reaction. Conservative results were further ensured with the following input: • Fuel rod input based on the maximum fuel temperature at the given power, • The hot spot power factor was equal to or greater than the design linear heat rate, • Uncertainties were applied to the initial operating conditions in the limiting direction.

Reference

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," S. L. Davidson (Ed.), July 1985.

ENCLOSURE 8

POINT BEACH NUCLEAR PLANT
LICENSE AMENDMENT REQUEST 241
ALTERNATIVE SOURCE TERM

LIST OF COMMITMENTS

LIST OF COMMITMENTS

NEW COMMITMENTS

Commitments for plant modifications and procedure changes are made in support of this application. Detailed descriptions of these new commitments are included in Enclosure 1.

- FPL Energy Point Beach will modify the PBNP control room (CR) radiation shielding to ensure CR habitability requirements are maintained. This modification is scheduled to be completed following Nuclear Regulatory Commission (NRC) approval of this license amendment request (LAR) 241 "Alternative Source Term" at the Unit 1 (2010) refueling outage.

- FPL Energy Point Beach will modify the containment spray (CS) and residual heat removal (RHR) systems to provide throttling capability of CS and RHR during the emergency core cooling system (ECCS) recirculation phase. These modifications will be completed on a unit specific basis at the next Unit 1 (2010) and Unit 2 (2009) refueling outages.

- FPL Energy Point Beach will revise PBNP emergency operating procedures (EOPs) to direct continued CS while on sump recirculation, if containment radiological conditions and/or core damage indicates it is required. These procedure changes will be implemented following NRC approval of this LAR 241 "Alternative Source Term" and following the completion of each unit specific installation of the CS and RHR system modifications to provide throttling capability during the ECCS recirculation phase.

- FPL Energy Point Beach will modify control room emergency filtration system (CREFS) to create a new alignment for the accident mode that provides a combination of filtered outside air and filtered recirculation air. The modifications will include redundancy for all CREFS active components that must reposition from their normal operating position, and auto-start capability on loss of offsite power in conjunction with a Containment Isolation or High Control Room Radiation signal from a diesel generator supplied source for the CREFS fans required for the new system alignment. This modification will be completed following NRC approval of this LAR 241 "Alternative Source Term" during the second site refueling outage that completes installation of the CS and RHR system modifications to provide throttling capability during the ECCS recirculation phase, thus completing installation for both units.

- FPL Energy Point Beach will modify the primary auxiliary building (PAB) ventilation system (VNPAB) to ensure redundancy of active components needed to operate the PAB filter and stack fans. VNPAB components credited for AST will be upgraded to an augmented quality status. No credit is taken by AST for the PAB charcoal filters. FPL Energy Point Beach will revise PBNP EOPs to address starting the VNPAB fans.

REVISED AND COMPLETED COMMITMENTS

- This application completes the following commitment for PBNP in response to Generic Letter (GL) 2003-01, "Control Room Habitability," June 12, 2003 (ADAMS Accession No. ML031620248) made by the Nuclear Management Company (NMC) LLC, the former license holder for PBNP, as revised in letter dated January 30, 2008 (ADAMS Accession No. ML080310556).

FPLE-PB 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 the end of 1Q09. As part of this submittal, the post-accident reliance on KI [potassium iodide] for control room staff will be addressed.

FPL Energy Point Beach submitted LAR 241, "Alternative Source Term," by letter 2007-0040 dated October 1, 2007 (ADAMS Accession No. ML072850206). By letter 2008-0004, dated January 30, 2008 (ADAMS Accession No. ML080310556), LAR 241 was withdrawn to resolve technical concerns related to the modifications associated with the AST. In the withdrawal letter, FPL Energy Point Beach revised the commitment date for GL 2003-01 from October 1, 2007, to prior to the end of the first quarter of 2009. This letter resubmits the AST LAR, using radiological analyses at a core power level equal to 1800 MWt with 0.6% uncertainties (analyzed core power of 1811 MWt) for both units, and satisfies the GL 2003-01 commitment discussed above.

- The regulatory commitment provided in response to GL 2003-01 dated May 1, 2007 (Adams Accession No. ML071210471) stating "NMC will provide Technical Specification changes to reference an acceptable surveillance methodology (and plans for any associated plant modifications to the control room envelope) to support requested information in GL 2003-01, Item (c), for PBNP no later than 180 days following NRC approval of TSTF-448." is revised for the AST as follows:

  FPL Energy Point Beach will submit a LAR addressing Control Room habitability surveillance methodology in accordance with TSTF-448, as modified by TSTF-508, within 60 days of approval of the AST LAR.

- As a part of PBNP's response to NUREG-0737, Item III.D.3.4 "Control Room Habitability Requirements," documented in NRC SER dated August 10, 1982 (not available in ADAMS), PBNP committed to the placement of portable lead shielding to limit radiation exposure through the control room doors and windows. Following installation of the permanent shielding modification for the control room that replaces the temporary shielding, this commitment will be eliminated.

- As a part of PBNP's application dated July 24, 2005, incorporating a PBNP reactor vessel head drop (RVHD) accident analysis and subsequent NRC SER, dated September 23, 2005 (Adams Accession No. ML052560089), PBNP committed to a reactor shutdown time of 100 hours prior to lifting the reactor vessel head. Following NRC approval of this amendment request for the RVHD radiological analysis using the AST methodology, this commitment will be eliminated, as the proposed RVHD radiological analysis using the AST methodology assumes the reactor head is lifted immediately following reactor shutdown.