September 28, 2007

Mr. Michael Balduzzi
Sr. Vice President, Regional Operations NE
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: PALISADES PLANT - ISSUANCE OF AMENDMENT RE: ALTERNATIVE RADIOLOGICAL SOURCE TERM (TAC NO. MD3087)

Dear Mr. Balduzzi:

The Commission has issued the enclosed Amendment No. 226 to Renewed Facility Operating License No. DPR-20 for the Palisades Nuclear Plant (PNP). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 25, 2006, as supplemented by letters dated June 15, September 7, September 20, and September 21, 2007.

The amendment provides the TS changes and evaluations of the radiological consequences of design basis accidents for implementation of a full-scope alternative source term methodology.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Registe*r notice.

Sincerely,

/RA/

Mahesh L. Chawla, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures:
1. Amendment No. 226 to DPR-20
2. Safety Evaluation

cc w/encls: See next page

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Sr. Vice President, Regional Operations NE
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ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 226
Renewed License No. DPR-20

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee), dated September 15, 2006, as supplemented by letters dated June 15, September 7, September 20, and September 21, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to the license amendment and Paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-20 is hereby amended to read as follows:

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 226, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Travis L. Tate, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Renewed Facility Operating License
and Technical Specifications

Date of Issuance: September 28, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 226

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Replace the following page of the Renewed Facility Operating License No. DPR-20 with the attached revised page. The changed area is identified by a marginal line.

REMOVE	INSERT
Page 3	Page 3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
1.1-3	1.1-3

(1) Pursuant to Section 104b of the Act, as amended, and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) ENP to possess and use, and (b) ENO to possess, use and operate, the facility as a utilization facility at the designated location in Van Buren County, Michigan, in accordance with the procedures and limitation set forth in this license;

(2) ENO, pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess, and use source and special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use byproduct, source, and special nuclear material as sealed sources for reactor startup, reactor instrumentation, radiation monitoring equipment calibration, and fission detectors in amounts as required;

(4) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration, or associated with radioactive apparatus or components; and

(5) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operations of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act; to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) ENO is authorized to operate the facility at steady-state reactor core power levels not in excess of 2565.4 Megawatts thermal (100-percent rated power) in accordance with the conditions specified herein.

(2) The Technical Specifications contained in Appendix A, as revised through Amendment No. 226, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) ENO shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SERs dated 09/01/78, 03/19/80, 02/10/81, 05/26/83, 07/12/85, 01/29/86, 12/03/87, and 05/19/89 and subject to the following provisions:

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 226 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

ENTERGY NUCLEAR OPERATIONS, INC.

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By application dated September 25, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML062830385), as supplemented by letters dated June 15 (ADAMS Accession No. ML071700698), September 7 (ADAMS Accession No. ML072540784), September 20, (ADAMS Accession No. ML072680173), and September 21, 2007 (ADAMS Accession No. ML072640608), Nuclear Management Company, LLC (the licensee at the time of submittal), requested an amendment to fully implement an alternative source term (AST) methodology at Palisades Nuclear Plant (PNP). The amendment provides the Technical Specification (TS) changes and evaluations of the radiological consequences of design-basis-accidents (DBAs) for implementation of a full-scope AST pursuant to Title 10 of the *Code of Federal Regulations* Part 50, Section 67 (10 CFR 50.67) and uses the methodology described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

The licensee's supplements dated June 15, September 7, September 20, and September 21, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 27, 2007 (72 FR 8804).

2.0 REGULATORY EVALUATION

The NRC staff evaluated the radiological consequences of affected DBAs for implementation of the AST methodology at PNP as proposed by the licensee against the dose criteria specified in 10 CFR 50.67(b)(2). These criteria are: 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release, 25 rem TEDE at the outer boundary of the low-population zone (LPZ) for the duration of the postulated fission product release, and 5 rem TEDE for access and occupancy of the control room (CR) for the duration of the postulated fission product release.

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements from which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183 and Standard Review Plan (SRP) Section 15.0.1. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183. The NRC staff's evaluation is based upon the following regulatory codes, guides, and standards:

- 10 CFR 50.67, "Accident Source Term."
- 10 CFR Part 50, Appendix A, "General Design Criterion for Nuclear Power Plants": GDC 19, "Control room."
- RG 1.23, "Meteorological Monitoring Programs For Nuclear Power Plants," Rev. 1, March 2007.
- RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Rev. 3, June 2001.
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Rev. 1, November 1982.
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Rev. 0, July 2000.
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Rev. 0, June 2003.
- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Rev. 0, May 2003.
- NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," May 1985.
- NUREG-0800, "Standard Review Plan," Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases," Rev. 3, March 2007.
- NUREG-0800, "Standard Review Plan," Section 6.4, "Control Room Habitability Systems," Rev. 3, March 2007.
- NUREG-0800, "Standard Review Plan," Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Rev. 4, March 2007.
- NUREG-0800, "Standard Review Plan," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000.
- NUREG-0800, "Standard Review Plan," Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Rev. 2, July 1981.

- NUREG-0933, Supplement 25, “A Prioritization of Generic Safety Issues,” June 2001.

- NUREG-0737, “Clarification of TMI [Three Mile Island] Action Plan Requirements,” November 1980.

- NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” July 1980.

- NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants,” 1979.

- NUREG/CR-4102, “Air Currents Driven by Sprays in Reactor Containment Buildings.”

- NUREG/CR-6331, “Atmospheric Relative Concentrations in Building Wakes,” Revision 1, May 1997.

- NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” February 1995.

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences of Design Basis Accidents

As stated in RG 1.183, Regulatory Position 5.2, the DBAs addressed in the appendices were selected from accidents that may involve damage to irradiated fuel. RG 1.183 does not address DBAs with radiological consequences based on TS limited primary or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in RG 1.183 should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST.

The licensee performed analyses for the full implementation of the AST, in accordance with the guidance in RG 1.183, and Section 15.0.1 of the SRP. The licensee performed AST analyses for the pressurized-water reactor (PWR) DBAs identified in RG 1.183 that could potentially result in significant CR and offsite doses. These include the loss-of-coolant accident (LOCA), the main steam line break accident (MSLB), the steam generator tube rupture (SGTR) accident, the control rod ejection accident (CREA) and the fuel-handling accident (FHA). In addition, the licensee performed analyses for the small line break outside containment (SLBOC) and the spent fuel cask drop (SFCD) accident, neither of which are specifically addressed in RG 1.183. For the SLBOC, the licensee used applicable guidance from NUREG-0800, “Standard Review Plan,” Section 15.6.2, “Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment,” Rev. 2, July 1981, with appropriate modifications to maintain the intent of RG 1.183. For the SFCD accident, the licensee followed the applicable requirements of RG 1.183 Appendix B, which describes assumptions for the evaluation of the radiological consequences of an FHA.

Section 14.7 of the PNP Final Safety Analysis Report (FSAR) discusses the consequences of decreased reactor coolant flow events, including the reactor coolant pump (RCP) seized rotor event. Section 14.7.2.6 concludes that for the RCP seized rotor event, there is a 95-percent confidence that departure from nucleate boiling (DNB) is not expected to occur and that no fuel failures are expected. RG 1.183, Appendix G, “Assumptions for Evaluating the Radiological

Consequences of a PWR Locked Rotor Accident," states that, "If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment." Therefore, the licensee asserts and the NRC staff agrees that, the inclusion of the RCP seized rotor event is not required for the full implementation of the AST at PNP.

The core inventory calculations used in the evaluation of the DBA radiological analyses for the AST amendment were performed assuming a power level of 2703 megawatts thermal (MWt). The primary coolant system (PCS) and secondary side activities are also based on the core inventories evaluated at 2703 MWt. All other thermal-hydraulic calculations are based on the current licensed power level of 2565.4 MWt including the current licensed calorimetric measurement uncertainty of 0.5925-percent for an analysis power level of 2580.6 MWt. Therefore, the power level of 2703 MWt applies to the core inventory calculations and does not apply to any other power related aspects of the AST submittal. This distinction in the use of power levels in the AST analyses was described to the NRC staff in a letter dated June 15, 2007. The use of the power level of 2703 MWt for the AST core inventory calculations is not intended to imply that AST calculations are valid for power levels not in excess of the currently licensed power of 2565.4 MWt. The use of 2703 MWt for the determination of the core inventories used in the AST DBA analyses bounds the current licensed core thermal power level of 2,565.4 MWt and is, therefore, acceptable to the NRC staff for use in the full implementation of the AST at PNP.

The licensee has proposed a full implementation of the AST as defined in RG 1.183. The licensee has determined that the current Technical Information Document (TID) 14844, AEC [Atomic Energy Commission], 1962, "Calculation of Distance Factors for Power and Test Reactors Sites," accident source term will remain the licensing basis for equipment qualification (EQ).

Regulatory Position 6 of RG 1.183 states that the NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted and that until such time as this generic issue is resolved, licensees may use either the AST or the TID-14844 assumptions for performing the required EQ analyses. This issue has been resolved as documented in a memo dated April 30, 2001 (ADAMS Accession No. ML011210348) and in NUREG-0933, Supplement 25, June 2001 (ADAMS Accession No. ML012190402). As stated in the conclusion to Generic Issue 187, "The NRC staff concluded that there was no clear basis for back-fitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement. Licensees should be aware, however, that a more realistic source term would potentially involve a larger dose for equipment exposed to sump water for long periods of time. Longer term equipment operability issues associated with severe fuel damage accidents, (with which the AST is associated) could also be addressed under accident management or plant recovery actions as necessary."

Therefore, in consideration of the cited references, the NRC staff finds that it is acceptable for the TID-14844 accident source term to remain the licensing basis for EQ at PNP.

In a letter dated June 15, 2007, the licensee provided additional information confirming that the EQ analyses support power levels of 2580.6 MWt which represents the licensed power level of 2565.4 MWt with a measurement uncertainty of 0.5925-percent. There is no relationship between the AST core inventory calculation power level of 2703 MWt and the EQ power level. The use of the power level of 2703 MWt for the AST core inventory calculations is not intended to imply that the EQ calculations are valid for power levels other than the currently licensed power level of 2565.4 MWt.

RG 1.183, Regulatory Position 4.3, Other Dose Consequences, states that, "The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE."

The licensee has proposed that the source term described in TID 14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactors Sites," should remain the licensing basis for NUREG-0737 evaluations other than CR habitability envelope (CRHE) and has made reference to the resolution of Generic Issue 187 as discussed in relation to the EQ dose evaluations. The licensee performed the NUREG-0737 post-accident access evaluations considering a 30 day duration for vital area dose rates. The licensee asserts and the NRC staff agrees that, based on the resolution of Generic Issue 187, for exposure to containment atmosphere, the TID-14844 source term and the AST will produce similar integrated doses; and that for exposure to sump water, the integrated doses calculated with the AST will only exceed those calculated with TID-14844 after 42 days for a PWR. In addition, the requirements to maintain post-accident sampling system capability at PNP were eliminated in Amendment No. 193. The licensee asserts, and the NRC staff agrees, that there would be no discernable risk reduction associated with the reconstruction of the post-accident access doses using the AST. Therefore, the NRC staff finds that it is acceptable for the TID-14844 accident source term to remain the licensing basis for the NUREG-0737 analyses at PNP.

In a letter dated June 15, 2007, the licensee confirmed that the NUREG-0737 analyses support power levels of 2580.6 MWt, which represents the licensed power level of 2565.4 MWt with a measurement uncertainty of 0.5925-percent. There is no relationship between the AST core inventory calculation power level of 2703 MWt and the NUREG-0737 power level. The use of the power level of 2703 MWt for core inventory calculations is not intended to imply that the NUREG-0737 dose calculations are valid for power levels not in excess of the currently licensed power level of 2565.4 MWt.

A full implementation of the AST is proposed for PNP. Therefore, to support the licensing and plant operation changes discussed in the license amendment request (LAR), the licensee analyzed the following accidents employing the AST as described in RG 1.183.

1. Loss of Coolant Accident (LOCA)

2. Fuel Handling Accident (FHA)

3. Main Steam Line Break (MSLB) Accident

4. Steam Generator Tube Rupture (SGTR) Accident

5. Small Line Break Outside Containment (SLBOC)
6. Control Rod Ejection Accident (CREA)

7. Spent Fuel Cask Drop (SFCD)

The DBA dose consequence analyses evaluated the integrated TEDE dose at the EAB for the worst 2-hour period following the onset of the accident. The integrated TEDE doses at the outer boundary of the LPZ and the integrated dose to a PNP CR operator were evaluated for the duration of the accident. The dose consequence analyses for the AST were performed for the licensee by Numerical Applications Inc. (NAI). The dose consequence analyses were performed using the RADTRAD-NAI computer code developed by NAI. The RADTRAD-NAI code is based on the RADTRAD code, which is used by the NRC staff for the performance of confirmatory dose consequence analyses. NRC sponsored the development of the radiological consequence computer code, "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03, as described in NUREG/CR-6604. The RADTRAD code estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff performed independent confirmatory dose evaluations using the RADTRAD computer code. The results of the evaluations performed by the licensee, as well as the applicable dose acceptance criteria from RG 1.183, are shown in Table 1.

RG 1.183, Regulatory Position 3.1, "Fission Product Inventory", states that "The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS [Emergency Core Cooling System] evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP." In accordance with RG 1.183 the licensee generated the core and worst case fuel assembly radionuclide inventories for use in determining source term inventories using the ORIGEN code version 2.1, Oak Ridge National Laboratory, CCC-371, "RSICC Computer Code Collection – ORIGEN 2.1," May 1999. The inventories, consisting of the curie levels for 107 dose significant isotopes at end of fuel cycle, formed the source term input for the RADTRAD-NAI dose evaluations.

As stated in RG 1.183, the release fractions associated with the light-water reactor (LWR) core inventory released into containment for the DBA LOCA and non-LOCA events have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 megawatt days per metric ton of uranium (MWD/MTU) provided that the maximum linear heat generation rate does not exceed 6.3 kilowatt per foot (kW/ft) peak rod average power for burnups exceeding 54,000 MWD/MTU.

Subsequent to the AST license amendment request, the licensee identified that expected fuel rod power levels were beyond the applicability of credited fission-product gap inventories. Specifically, footnote 11 of Table 3 within RG 1.183 restricts the use of the gap fractions to fuel rods whose maximum linear heat generation rate does not exceed 6.3 kW/ft for burnups exceeding 54,000 MWD/MTU. The original AST LAR included an assumption that no fuel rod would exceed the footnote 11 applicability criteria. However, the licensee recently determined, based upon core reload depletion calculations, that this criteria would be violated for a small number of fuel rods in current and future cycles.

Fuel rods operating beyond 6.3 kW/ft at a high burnup may experience significant fission gas release (into the rod plenum). The main concern is that the empirical database of fission gas measurements on this type of power history is limited. In Enclosure 3 of the supplement sent on September 7, 2007, the licensee provided an alternative method for validating the RG 1.183 Table 3 gap fractions used in the AST LAR. In addition to maintaining the TS peaking factor limit of 2.04, the licensee would double the gap fraction for the fuel rods which exceed the 54,000/6.3 criterion. In past AST reviews, the NRC staff has accepted doubling the RG 1.183 gap fraction to compensate for the uncertainty associated with the gap fraction.

In the September 7, 2007, letter, the licensee proposed fuel management restrictions which limit the overall number of fuel rods which violate the 54,000/6.3 criterion to fewer than 21 in any one assembly and fewer than 20 assemblies in any core design. Further, two new applicability criteria, (1) less than 6.7 kW/ft and (2) less than 58,500 MWD/MTU, are imposed on any fuel rod which violates the 54,000/6.3 criterion. Based upon engineering judgement and past practice, the NRC staff finds the power level and burnup limits acceptable to ensure gap fractions remain below 2.0 times RG 1.183 Table 3 gap fractions.

The licensee also proposed a fuel management restriction which would preserve the original assumptions within the AST dose calculation. This restriction would ensure that a sufficient number of rods operating at powers below the TS peaking factor were available to offset the doubling of the gap fraction. Since a rod average source term is multiplied by the core maximum peaking factor, margin exists in any fuel rod whose actual peaking factor remains below this power peaking limit. The NRC staff had concerns with crediting administrative controls (e.g., fuel management guidelines), more restrictive than the TS Core Operating Limits Report (COLR) limit, to preserve a key assumption within the dose calculation. In a letter dated September 21, 2007, the licensee has committed to updating the COLR by adding the 5 rod power restrictions from the September 7, 2007, letter. Based upon these COLR restrictions, the NRC staff finds the fission-product gap fractions used in the AST dose calculations acceptable.

The licensee used committed effective dose equivalent (CEDE) and effective dose equivalent dose conversion factors (DCFs) from Federal Guidance Reports (FGRs) 11 and 12 to determine the TEDE dose as is required for AST evaluations. The use of ORIGEN and DCFs from FGR-11 and FGR-12 is in accordance with RG 1.183 guidance and is, therefore, acceptable to the NRC staff.

3.1.1 Loss-of-Coolant Accident (LOCA)

The radiological consequence design basis LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the primary reactor coolant system (RCS) piping. The accident scenario assumes the deterministic failure of the ECCS to provide adequate core cooling that results in a significant amount of core damage as specified in RG 1.183. This general scenario does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design basis transient analyses.

The LOCA considered in this evaluation is a complete and instantaneous circumferential severance of the primary RCS piping, which would result in the maximum fuel temperature and primary containment pressure among the full range of LOCAs. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. The fission product release is assumed to occur in phases over a 2-hour period.

When using the AST for the evaluation of a design basis LOCA for a PWR, it is assumed that the initial fission product release to the containment will last for 30 seconds and will consist of the radioactive materials dissolved or suspended in the RCS liquid. After 30 seconds, fuel damage is assumed to begin and is characterized by clad damage that releases the fission product inventory assumed to reside in the fuel gap. The fuel gap release phase is assumed to continue until 30 minutes after the initial breach of the RCS. As core damage continues, the gap release phase ends and the early in-vessel release phase begins. The early in-vessel release phase continues for the next 1.3 hours. The licensee used the LOCA source term release fractions, timing characteristics, and radionuclide grouping as specified in RG 1.183 for evaluation of the AST.

In the evaluation of the LOCA design basis radiological analysis the licensee included dose contributions from the following activity release pathways:

- Containment leakage
- Engineered safety feature (ESF) system leakage
- ESF system back-leakage into the safety injection refueling water tank (SIRWT)

The licensee included the following DBA LOCA dose contributors to the CRHE analysis:

- Contamination of the CR atmosphere by released activity
- Plume shine from released activity
- Shine from the containment, purge lines, SIRWT and CR filter loading

3.1.1.1 Source term

The licensee followed all aspects of the guidance outlined in RG 1.183, Regulatory Position 3, regarding the fission product inventory, release fractions, timing, radionuclide composition and chemical form in the evaluation of the LOCA.

RG 1.183, Appendix A, Regulatory Position 2 states, "If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95-percent cesium iodide (CsI), 4.85-percent elemental iodine, and 0.15-percent organic iodine. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event (e.g., radiolysis products). With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form."

The licensee asserts that the sump pH is controlled to a value greater than 7.0 based on the addition of a suitable buffer. In the application, the licensee committed that during the 2007 fall refueling outage at PNP, they would implement a buffer program to maintain a pH of 7.0 - 8.0 post-LOCA with recirculation. Therefore, in accordance with RG 1.183, Appendix A, Regulatory

Position 2, the licensee has assumed that the chemical form of the radioiodine released to the containment is 95-percent CsI, 4.85-percent elemental iodine and 0.15-percent organic iodine. Additionally, as prescribed in RG 1.183, the licensee assumed that, with the exception of elemental and organic iodine and noble gases, all fission products are in particulate form.

3.1.1.2 Assumptions on transport in the primary containment

3.1.1.2.1 Containment mixing, natural deposition and leak rate

In accordance with RG 1.183, the licensee assumed that the activity released from the fuel is mixed instantaneously and homogeneously throughout the free air volume of the containment. The licensee used the core release fractions and timing as specified in RG 1.183 with the termination of the release into containment set at the end of the early in-vessel phase.

The licensee credited the reduction of airborne radioactivity in the containment by natural deposition. The licensee credited an elemental iodine natural deposition removal coefficient of 2.3 hr^{-1} for the duration of the accident. The licensee cited EA-PAH-91-06, Revision 2, Fission Product Removal Coefficients for Design Basis Radiological Consequence Analyses, September 2006, as the reference for the elemental iodine natural deposition removal coefficient of 2.3 hr^{-1}. In a letter dated June 15, 2007, the licensee provided additional information describing the technical basis for the elemental iodine removal coefficient of 2.3 hr^{-1}. The licensee used conservative assumptions to calculate a total wetted containment surface area, from the activation of containment sprays, of 233,343 ft^2. The licensee used the mass transfer coefficient of 4.9 meters per hour as suggested in SRP 6.5.2 in order to conservatively bound the available experimental data. Using this mass-transfer coefficient, the wetted surface area as discussed, and the net containment free volume of 1,64E+06 ft^3, the licensee calculated an elemental iodine removal coefficient of 2.3 hr^{-1} for wall deposition.

The licensee assigned an aerosol natural deposition removal coefficient of 0.1 hr^{-1} based on the Industry Degraded Core Rulemaking Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983. SRP 6.5.2 does not specify the summation of the natural deposition and spray removal coefficients for aerosols. Therefore, for conservatism the licensee did not add these removal coefficients together during the time of spray operation. The licensee incorporated an aerosol natural deposition removal coefficient of 0.1 hr^{-1} into the dose analysis starting at 10 hours post accident, after the termination of the containment spray system, and continuing for the duration of the accident. The licensee did not credit the removal of organic iodine by natural deposition. The NRC staff finds the licensee's methods to model both elemental and aerosol iodine natural deposition in the containment to be conservative and, therefore, acceptable.

RG 1.183, Regulatory Position 3.7 states that, "The primary containment should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50-percent of the technical specification leak rate." Accordingly, the licensee assumed a containment leak rate of 0.10 weight percent per day for the first 24 hours, after which the containment leak rate is reduced to 0.05 weight percent per day for the duration of the accident.

The licensee did not consider the effects of online containment purging in the LOCA analysis. In a letter dated June 15, 2007, the licensee provided additional information describing the basis for not including the effects of online containment purging in the LOCA analysis. The licensee asserts that PNP does not perform routine containment purges and that post LOCA hydrogen control is not a design basis issue since the hydrogen recombiners have been eliminated from the PNP design basis. Continuous containment venting can be established by the removal of the clean waste receiver disk, RUD-1018, through control valves CV-1064 and CV-1065. The licensee asserts that online continuous venting through these valves is acceptable since TS stroke time surveillance testing assures that these valves, as well as other containment isolation valves, will be isolated in less than 30 seconds after the initiation of a containment isolation signal. The AST recognizes that the onset of fuel failure will not occur instantaneously. The TS surveillance testing assures that the control valves CV-1064 and CV-1065 will automatically close prior to the onset of any significant fuel failure and, therefore, the exclusion of the effects of releases through this pathway for the AST LOCA analysis is acceptable to the NRC staff.

3.1.1.2.2 Containment spray assumptions

SRP Section 6.5.2, III,1,c states, "The containment building atmosphere may be considered a single, well-mixed space if the spray covers regions comprising at least 90-percent of the containment building space and if a ventilation system is available for adequate mixing of any unsprayed compartments." RG 1.183, Appendix A, Regulatory Position 3.3 dropped the reference to the availability of a ventilation system to ensure adequate mixing by stating that, "The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90-percent of the volume and if adequate mixing of unsprayed compartments can be shown." The licensee references Amendment No. 31 and states that, "Per the current licensing basis, there is at least 90-percent spray coverage of the containment; therefore, the containment is treated as a single well mixed volume." In a letter dated June 15, 2007, the licensee provided additional information regarding the assumption of a single well mixed containment volume. The licensee states that adequate mixing is assured based on the operation of containment sprays, thermally driven natural convection currents, and induced flow from the blowdown of the RCS into the containment. The containment air cooler fans, as described in FSAR Section 6.3, are not credited in the LOCA dose analysis, but would provide mixing of the containment atmosphere to prevent local accumulation of combustible gases and improve the effectiveness of the iodine removal rate of the containment sprays. The licensee also cited NUREG/CR-4102, "Air Currents Driven by Sprays in Reactor Containment Buildings," as providing indication that spray induced mixing is substantial relative to the mixing that results from fan operation. The NRC staff finds that the assumption that the containment can be considered a single well mixed volume is acceptable for PNP based on the extent of spray coverage, the mixing that will occur from the operation of containment sprays, LOCA hydraulic forces and thermal convection currents, as well as the existence of the uncredited containment air cooler fans.

The licensee performed RADTRAD-NAI analyses to determine the spray time required to reach the maximum acceptable elemental iodine decontamination factor (DF) of 200 as prescribed in SRP Section 6.5.2. The analyses included a case to determine the maximum containment atmosphere elemental iodine concentration and the maximum amount of aerosol in the containment atmosphere. This case was performed with no containment spray, no iodine surface deposition, no decay and no containment leakage. A second case was performed with containment spray based on the maximum elemental iodine concentration as determined in the first case. The second case used similar assumptions of no iodine surface deposition, no decay and no containment leakage and determined that an elemental iodine DF of 200 would be reached at a time greater than 2.515 hours. Therefore, in accordance with the applicable guidance, the licensee used an elemental iodine spray removal rate constant of 4.8 per hour for a time period of 2.525 hours after which no further reduction in elemental iodine from the operation of containment sprays iodine was credited.

A third case was performed to determine the time required to reach a DF of 50 for aerosol deposition based on the maximum aerosol mass from the first case. This case assumed containment spray actuation at 1 minute, credited aerosol deposition and assumed no decay or containment leakage. As a result of this evaluation the licensee determined that an aerosol DF of 50 would be reached at a time greater than 3.385 hours. Therefore, in accordance with the applicable guidance, the licensee used an aerosol iodine spray removal rate constant of 1.8 per hour for a time period of 3.385 hours after which the aerosol iodine spray removal rate constant was decreased to a value of 0.18 per hour until the cessation of containment spray operation at 10 hours post LOCA.

The licensee's evaluation of the reduction of elemental and aerosol iodine airborne activity in the containment atmosphere as a result of the operation of the containment spray system used conservative assumptions, as prescribed by the applicable regulatory guidance, and is, therefore, acceptable to the NRC staff for use in the AST LOCA analysis.

3.1.1.3 Assumptions on Engineered Safety Feature (ESF) System Leakage

To evaluate the radiological consequences of ESF leakage the licensee used the deterministic approach as prescribed in RG 1.183. This approach assumes that except for the noble gases, all of the fission products released from the fuel mix instantaneously and homogeneously in the containment sump water. Except for iodine, all of the radioactive materials in the containment sump are assumed to be in aerosol form and retained in the liquid phase. As a result, the licensee assumed that the fission product inventory available for release from ECCS leakage consists of 40-percent of the core inventory of iodine. This amount is the combination of 5-percent released to the containment sump water during the gap release phase and 35-percent released to the containment sump water during the early in-vessel release phase. This source term assumption is conservative in that 100-percent of the radioiodines released from the fuel are assumed to reside in both the containment atmosphere and in the containment sump concurrently.

ECCS leakage develops when ESF systems circulate containment sump water outside containment and leaks develop through packing glands, pump shaft seals and flanged connections. For the LOCA analysis of ESF leakage, the licensee used a value of 0.4 gallons per minute (gpm). As specified in RG 1.183, Appendix A, Item 5.2, this value represents two times the TS allowable value of 0.2 gpm. The licensee assumed that ESF leakage begins at 19 minutes post-LOCA, which is the earliest time that recirculation flow is projected to begin. The AST LOCA analysis assumes that the ESF leakage will persist for the 30-day duration of the accident.

3.1.1.3.1 Assumptions on ESF system leakage to the auxiliary building

RG 1.183, Appendix A, Regulatory Position 5.5, states that, "If the temperature of the leakage is less than 212 °F or the calculated flash fraction is less than 10-percent, the amount of iodine that becomes airborne should be assumed to be 10-percent of the total iodine activity in the leaked fluid." The licensee calculated a flash fraction of 3-percent based on the temperature of the containment sump liquid at the time that recirculation begins. In accordance with RG 1.183, the licensee used a flash fraction of 10-percent to evaluate the contribution due to ESF leakage into the auxiliary building for the LOCA analysis. In accordance with RG 1.183, for ESF leakage into the auxiliary building, the licensee assumed that the chemical form of the released iodine is 97-percent elemental and 3-percent organic. The licensee did not credit a reduction of activity released to the auxiliary building as a result of dilution or holdup.

Per the current design basis, the licensee credited a 50-percent reduction of the ECCS leakage into the auxiliary building. In a letter dated June 15, 2007, the licensee provided additional information regarding the assumption of 50-percent reduction of the ECCS leakage into the auxiliary building. The ESF room ventilation is normally aligned to the stack. ESF room ventilation would be isolated on high radiation, however the radiation detectors, closure signal, dampers and duct work are not classified as safety systems. This issue was evaluated as part of the systematic evaluation program NUREG-0820, Topics IX-5, "Ventilation Systems," and XV-19, "Loss-of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary - Radiological Consequences." A condition of significant leakage from the ESF room was evaluated and the assumption of an iodine plate out factor of 2 was accepted.

The licensee asserts, and the NRC staff agrees, that for the DBA radiological analysis of significant leakage from the failure of the ESF room isolation dampers to close, an iodine plate out factor of 2 has been accepted as the PNP design basis and is, therefore, acceptable for use in the AST analysis. The licensee further states that since all stack releases are modeled as ground level releases and since the stack is closer to the CR normal intakes than the ESF room, the assumption of a stack release for ESF room leakage is conservative and, therefore, acceptable to the NRC staff for the AST evaluation.

3.1.1.3.2 Assumptions on ESF system back-leakage to the SIRWT

The licensee evaluated back-leakage to the SIRWT separately as a portion of the ESF leakage contribution to the LOCA dose. The licensee assumed the back-leakage to the SIRWT to be two times the maximum allowed total leakage of 2.2 gpm. Therefore, from the time of recirculation initiation until 2 hours into the event the licensee assumed the back-leakage to the SIRWT to be 4.4 gpm. At 2 hours post-LOCA, the licensee credited operator action to cross-tie the low pressure safety injection (LPSI) suction headers to eliminate back-leakage through the SIRWT discharge lines. After 2 hours, the SIRWT back-leakage is reduced to two times the maximum allowed recirculation line leakage of 0.025 gpm. Therefore, from 2 hours post-LOCA and for the remainder of the event the licensee assumed the SIRWT back-leakage to be 0.05 gpm.

As stated on page 9 of NAI-1149-014, Revision 3, Palisades Design Basis AST MHA/LOCA Radiological Analysis, "Pre-staging of a temporary modification to accomplish the cross-tie as well as verification of the timing of the operator action is required to support this assumption. A modification to reduce the back-leakage through recirculation valves CV-3027 and CV-3056 is required to support the recirculation line leakage assumption."

The licensee has made commitments to: replace the ECCS pump minimum flow recirculation isolation valves (see Section 3.5.2.1 of this SE); modify plant emergency operating procedures to allow the cross-tie of the low pressure safety injection suction piping post LOCA following recirculation (see Section 3.5.2.3 of this SE); and to conduct any necessary testing (see Section 3.5.3 of this SE), following the implementation of the plant modifications described above, to validate that the modified plant configuration supports the assumptions used in the dose consequence analyses supporting the AST LAR.

The licensee's SIRWT leakage model accounts for the dose due to leakage of sump fluid back into the SIRWT, which subsequently is released to the environment through the SIRWT tank vent. As allowed by RG 1.183, Appendix A, Regulatory Position 5.6, the licensee applied site-specific conditions to credit holdup and dilution in the SIRWT. This evaluation is a complex process involving many variables. The sump fluid that leaks back into the SIRWT is assumed to mix with liquid in the SIRWT. The fraction of elemental iodine in the SIRWT is a function of the SIRWT pH and total iodine concentration. The elemental iodine in the SIRWT fluid is then assumed to enter the SIRWT air space as a function of the iodine partition coefficient (PC). The iodine PC is a function of the SIRWT liquid temperature. The iodine in the SIRWT air space is then available for release via the SIRWT vent.

The licensee asserts, and the NRC staff agrees, that based on the leakage rate and the size of the piping, the back-leakage would not reach the SIRWT until an extended period of time after recirculation begins. The licensee did not credit this time period for the purpose of accounting for radiological decay; however the licensee did credit this time period in the determination of the temperature of the back-leakage reaching the SIRWT.

As previously stated, the licensee calculated a flash fraction of approximately 3-percent based on the temperature of the containment sump liquid at the time that recirculation begins. The licensee has also determined that flashing will cease at approximately 4.9 hours into the event. Since the flashing fraction is very low and since flashing ceases at 4.9 hours, the licensee assumed that all of the back-leakage will condense within the pipe leading into the SIRWT and mix with the water inventory of the tank. The licensee confirmed this assumption using a GOTHIC model to determine the heat loss from the leaking fluid as it traveled to the SIRWT. The licensee's model indicates that the temperature of the ECCS fluid will cool to less than 212 °F prior to reaching the SIRWT. Given the small flashing fraction, the short time period involved before flashing would cease, and the results of the licensee's heat loss model, the NRC staff finds the assumption that all of the back-leakage will condense within the pipe leading into the SIRWT and mix with the water inventory of the tank, to be reasonable and acceptable for use in the calculation of the dose consequence of SIRWT back-leakage at PNP.

The licensee's evaluation also considered the effects of operator actions to partially refill the SIRWT post-LOCA. Water make-up to the SIRWT would likely have the following characteristics: a low pH, which would tend to increase the SIRWT iodine volatile fraction; a lower temperature relative to back-leaked sump fluid, which would tend to increase the SIRWT iodine PC; a high rate of addition relative to back-leakage rate, which would increase the SIRWT vent release rate; and a very low iodine concentration relative to the sump fluid, which would tend to decrease the SIRWT iodine concentration and therefore the iodine volatile fraction.

Due to the physical location of the SIRWT above the CR, there is a direct shine dose contribution to the CR operators as well as an airborne contribution from the SIRWT vent release. The direct dose to the CR from the activity in the SIRWT will be dependant on the initial volume of water in the SIRWT and the rate of back-leakage to the SIRWT. To account for the potentially nonconservative effects of self shielding, the licensee considered the possibility that the scenario in which all leaked back activity is retained in the tank may not bound the scenario in which, as a result of out-leakage, a minimum water level in the SIRWT exists for the duration of the event.

To assess the various conditions that could impact the SIRWT calculations, the licensee analyzed two sensitivity cases in addition to the design basis case, to ensure the limiting scenario has been considered for both the SIRWT vent release dose and direct shine dose contribution. The design basis case analyzed by the licensee assumes no SIRWT refill and that all activity leaked into the tank is retained (i.e., with no SIRWT out-leakage). The licensee performed two additional sensitivity cases. One case assumed a bounding SIRWT refill and no SIRWT out-leakage to maximize the SIRWT water level. This case was performed to ensure the SIRWT refill, which adds low pH water to the SIRWT and significantly increases the displacement vent flow during the addition but also provides significant dilution of iodine in the SIRWT, would not have a significant impact on SIRWT vent release dose.

The licensee evaluated a second case assuming no SIRWT refill and full SIRWT out-leakage which would minimize the SIRWT water level. This case was performed to ensure that full out-leakage, which decreases the total activity in the SIRWT but increases SIRWT activity concentration and provides less water for source self-shielding, does not have a significant impact on SIRWT shine dose. The sensitivity cases performed by the licensee demonstrated no significant difference in vent release leakage dose from the design basis case and that the SIRWT shine dose from the design basis SIRWT back-leakage case is bounding.

The licensee performed an extensive evaluation of the dose contribution from ESF back-leakage into the SIRWT. This evaluation included both the effects of the release of activity from the SIRWT vent and the direct shine contribution to the LOCA CR dose. The NRC staff has reviewed the licensee's evaluation and found it to be thorough and based on sound engineering judgements. In addition, the NRC staff finds that the licensee's evaluation meets the intent of all the applicable guidance regarding this contribution to LOCA dose. Therefore, the NRC staff finds the licensee's SIRWT back-leakage dose evaluation methodology to be acceptable for use in the AST license amendment.

3.1.1.4 Control Room Habitability

3.1.1.4.1 CR ventilation assumptions

As per regulatory guidance, the LOCA is assumed to occur concurrently with a loss of off site power (LOOP). Therefore, the licensee assumed that the initial airflow into the CR consists of 384.3 cubic feet per minute (cfm) of unfiltered air as a result of the loss of normal CR ventilation. This condition is assumed to persist for a period of 90 seconds after which the licensee credits both CR isolation and CR emergency mode ventilation. In the emergency mode, the CR ventilation consists of 1413.6 cfm of filtered makeup airflow through the emergency intake and 1413.6 cfm of filtered recirculation flow. For the LOCA CR habitability analysis the licensee assumed an unfiltered inleakage of 10 cfm after CR isolation. All CR unfiltered inleakage is assessed using the airborne concentrations at the normal CR intake. The licensee has committed to replace CR heating, ventilation, and air conditioning (HVAC) dampers D-1, D-2, D-8, D-9, D-15, and D-16 to support this assumption. Additionally, the licensee has committed to post modification and periodic tracer gas testing to confirm CR envelope (CRE) inleakage assumptions. The licensee credited the CR ventilation filter efficiencies, as applied to both the filtered makeup flow and the recirculation flow, as 99-percent for particulate activity, elemental iodine and organic iodine.

3.1.1.4.2 Direct shine dose evaluations

The total CR LOCA dose includes direct shine contributions from the following DBA-LOCA radiation sources:

- Contamination of the CR atmosphere by the intake and infiltration of the radioactive material contained in the radioactive plume released from the facility,
- Direct shine from the external radioactive plume released from the facility with credit for CR structural shielding,
- Direct shine from radioactive material in the containment with credit for both the containment and CR structural shielding,
- Radiation shine from radioactive material in systems and components inside or external to the CR envelope (e.g., activity in the containment purge lines, activity collected in the SIRWT and radioactive material buildup on the CR ventilation filters).

RG 1.196 defines the CRE as follows: “The plant area, defined in the facility licensing basis, that in the event of an emergency, can be isolated from the plant areas and the environment external to the CRE. This area is served by an emergency ventilation system, with the intent of maintaining the habitability of the CR. This area encompasses the CR, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident.”

The licensee evaluated the direct dose contribution from the contamination of the CR atmosphere by the intake and infiltration of the radioactive material contained in the radioactive plume released from the facility using the RADTRAD-NAI code as discussed previously. The licensee evaluated the remainder of the contributions to direct shine dose to the CR under LOCA conditions in two separate calculations. The first calculation, NAI-1149-024, Rev. 3, used the MicroShield 5 shielding code to evaluate dose contributions from the containment, the CR filters, external purge lines, and external cloud sources. The second calculation, NAI-1180-002, Rev. 2, used the QADMOD-GP shielding code to evaluate the direct shine dose to the CR from the SIRWT.

The licensee assumed, for the purposes of the direct dose contribution, that the unit of exposure, the roentgen, as determined in the shielding codes, is equivalent to the unit of dose to air, the rad, which in turn is conservatively assumed to numerically correspond to the units of rem for Deep-Dose Equivalent (DDE). DDE is then used in the summation required to determine the TEDE dose for which the 10 CFR 50.67 limits are based. The licensee assumed that all other sources of direct shine dose are negligible compared to the containment, CR filter, SIRWT, external purge lines, and external cloud sources. The licensee applied a CR occupancy factor of 1.0 for the first day, 0.6 from 1 to 4 days and 0.4 from 4 to 30 days to the calculated exposure, and therefore dose, for all cases considered. To evaluate the dose contribution from the external cloud the licensee assumed a source length of 1000 meters (m) in the MicroShield cases to approximate an infinite cloud. The use of 1000 m is consistent with the methodology of previously submitted AST analyses and is acceptable to the NRC staff. The licensee neglected the steel reinforcement bar (Rebar) in the concrete for shielding purposes. The licensee asserts that the effect of Rebar on the shielded dose calculation is not considered to be significant. Notwithstanding the significance of neglecting the presence of the Rebar in the shielding calculation, the exclusion of the effects of the Rebar will result in a higher shielded dose result, which is conservative and, therefore, acceptable to the NRC staff.

The SIRWT is constructed of aluminum, however, for conservatism the licensee neglected the presence of the bottom of the aluminum tank in the direct dose evaluation. The licensee made conservative assumptions regarding the depth of sand credited in the shielding analysis, which also resulted in conservatively decreasing the tank bottom elevation. Although the concrete pad upon which the SIRWT rests extends beyond the diameter of the base of the SIRWT, the licensee conservatively assumed that the pad exists only directly below the SIRWT.

The licensee evaluated the CR direct dose assuming that the receptor point is at an elevation 3 ft above the CR floor. The NRC staff considers this to be a reasonable assumption since it represents the height for the torso region of an average CR occupant. In addition, the PNP CR roof is a cellular structure that is not a uniform configuration. Modeling the CR roof cellular structure for shielding purposes is a complex problem and required the use of the QADMOD-GP shielding code to adequately account for the geometry involved. Because of the unique shielding of the CR roof cellular structure, the licensee has determined that a higher dose point, which would be closer to the SIRWT, does not necessarily yield a larger dose due to the competing effects of distance from the source and the amount of shielding a ray from the source passes through to the dose point.

The licensee conservatively neglected structures and components in the CR area in the shielded dose calculations. The licensee modeled the CR area as an open space with air as the only material between the elevations of 625 ft and 637 ft. The licensee conservatively assumed that none of the nuclides that enter the SIRWT during the post-LOCA period leave the SIRWT. The licensee accounted for decay, but for the shielding analysis the licensee conservatively neglected the transfer of nuclides from the SIRWT water to the tank atmosphere and subsequently to the environment.

The licensee used a modified version of QADMOD-GP to determine direct shine exposure to a dose point located in the CR from the SIRWT. The QADMOD-GP shielding code allows for shielding in three dimensions. To ensue an adequate accounting for the direct dose from the SIRWT the QADMOD-GP code was modified to allow for the cylindrical source to be represented with 99,999 source points rather than the standard 3000 points.

The licensee used conservative assumptions to evaluate the direct shine dose to the CR as a result of the LOCA. The results of the licensee's CR direct shine evaluations are included in Table 4 together with the CR habitability data and assumptions. The licensee conservatively included the CR direct shine as determined for the LOCA analysis, in all the other DBA CR habitability results, with the exception of the direct shine from the SIRWT which was only included in the LOCA CR habitability analysis. The licensee's assessment of the direct shine dose contribution to the CR habitability analysis for the AST LAR was performed following the applicable regulatory guidance and using sound engineering principals and is, therefore, acceptable to the NRC staff.

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 5 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the LOCA meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.2 Fuel-Handling Accident (FHA)

The FHA, as described in Section 14.19 of the PNP FSAR, consists of the drop of a single fuel assembly in the fuel-handling building (FHB) or inside containment. The FSAR description of the FHA specifies that all of the fuel rods in a single fuel bundle are damaged as a result of being dropped during fuel handling. In addition, a minimum water level of 22.5 ft is maintained above the damaged fuel assembly for both the containment and FHB release locations. For the FHA inside containment, the licensee assumed that the equipment hatch was open at the time of the accident. For the FHA in the FHB, the licensee analyzed three separate cases assuming 10-percent, 34-percent, and 50-percent of the released effluent is processed through the FHB filtration system. In accordance with RG 1.183, the licensee assumed that the release to the environment from the FHA occurs over a 2 hour period.

3.1.2.1 Source term

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged as a result of the postulated design basis FHA. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released as a result of the accident. The licensee performed a detailed analysis to ensure that the most restrictive case would be considered for the FHA dose consequence analysis. The process conducted by the licensee for determining the activities of the individual isotopes for the FHA was as follows:

1. The end of cycle plus 2 days of decay (39,300 MWD/MTU) activities for each isotope were extracted from the three core average burnup assembly ORIGEN cases.

2. For each isotope, the maximum activity was determined. This ensured that the range of specified enrichments was bounded.

3. A radial peaking factor of 2.04 was applied to the activities from Step 2.

4. The end of cycle plus two days of decay (58,900 MWD/MTU) activities for each isotope were extracted from the three peak burnup assembly ORIGEN cases.

5. For each isotope extracted in Step 4, the maximum activity was determined. This ensured that the range of specified enrichments was bounded.

6. For each isotope the maximum activity from Steps 3 and 5 was determined.

Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool (SFP) depending on their physical and chemical form. Following the guidance in RG 1.183, Appendix B, Regulatory Position 1.3 the licensee assumed that; the chemical form of radioiodine released from the fuel to the SFP consists of 95-percent CsI, 4.85-percent elemental iodine, and 0.15-percent organic iodine, the CsI released from the fuel is assumed to completely dissociate in the pool water, and because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This results in a final iodine distribution of 99.85-percent elemental iodine and 0.15-percent organic iodine. The licensee assumed that the release to the pool water and the chemical redistribution of the iodine species occurs instantaneously.

RG 1.183, Appendix B, Regulatory Position 2, allows an overall iodine DF of 200 for a water cover depth of 23 ft. The PNP FHA evaluation is based on a minimum water cover depth of 22.5 ft. In a letter dated June 15, 2007, the licensee provided additional information concerning the minimum water cover used in the FHA analysis. The licensee conservatively assumed that the FHA occurs at the location where the lowest water height exists which is the reactor cavity floor. The licensee stated that the likelihood of an FHA occurring on the reactor cavity floor is very small due to the fact that the fuel handling machine clearance above the floor is only a few inches, which would minimize the impact of a dropped assembly. To bound all possible events, the licensee used the smaller height of water cover while postulating the failure of all rods in an assembly. At this worst case location TSs ensure that the minimum water cover would be 22.5 ft.

For a water cover depth of 22.5 ft, the licensee calculated an elemental iodine DF of 252 and an overall iodine DF of 183.07 using guidance from RG 1.183 and from a technical paper entitled, "Evaluation of Fission Product Release and Transport for Fuel Handling Accident," G. Burley, 1971 (NRC legacy library accession number 8402080322). Consistent with RG 1.183, the licensee credited an infinite DF for the remaining particulate forms of the radionuclides contained in the gap activity. In accordance with RG 1.183, the licensee did not credit decontamination from water scrubbing for the noble gas constituents of the gap activity.

The licensee analyzed the FHA based on the fuel rod gap activity release fractions from RG 1.183, Regulatory Position 3, Table 3. RG 1.183 states that these release fractions have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54,000 MWD/MTU.

Subsequent to the AST license amendment request, the licensee identified that expected fuel rod power levels were beyond the applicability of credited fission-product gap inventories. Specifically, footnote 11 of Table 3 within RG 1.183 restricts the use of the gap fractions to fuel rods whose maximum linear heat generation rate does not exceed 6.3 kW/ft for burnups exceeding 54,000 MWD/MTU. The original AST LAR included an assumption that no fuel rod would exceed the footnote 11 applicability criteria. However, the licensee recently determined, based upon core reload depletion calculations, that this criteria would be violated for a small number of fuel rods in current and future cycles.

Fuel rods operating beyond 6.3 kW/ft at a high burnup may experience significant fission gas release (into the rod plenum). The main concern is that the empirical database of fission gas measurements on this type of power history is limited. In Enclosure 3 of the supplement sent on September 7, 2007, the licensee provided an alternative method for validating the RG 1.183 Table 3 gap fractions used in the AST LAR. In addition to maintaining the TS peaking factor limit of 2.04, the licensee would double the gap fraction for the fuel rods which exceed the 54,000/6.3 criterion. In past AST reviews, the NRC staff has accepted doubling the RG 1.183 gap fraction to compensate for the uncertainty associated with the gap fraction.

In the September 7, 2007, letter, the licensee proposed fuel management restrictions which limit the overall number of fuel rods which violate the 54,000/6.3 criterion to fewer than 21 in any one assembly and fewer than 20 assemblies in any core design. Further, two new applicability criteria, (1) less than 6.7 kW/ft and (2) less than 58,500 MWD/MTU, are imposed on any fuel rod which violates the 54,000/6.3 criterion. Based upon engineering judgement and past practice, the NRC staff finds the power level and burnup limits acceptable to ensure gap fractions remain below 2.0 times RG 1.183 Table 3 gap fractions.

The licensee also proposed a fuel management restriction which would preserve the original assumptions within the AST dose calculation. This restriction would ensure that a sufficient number of rods operating at powers below the TS peaking factor were available to offset the doubling of the gap fraction. Since a rod average source term is multiplied by the core maximum peaking factor, margin exists in any fuel rod whose actual peaking factor remains below this power peaking limit. The NRC staff had concerns with crediting administrative controls (e.g., fuel management guidelines), more restrictive than the TS Core Operating Limits Report (COLR) limit, to preserve a key assumption within the dose calculation. In letter dated September 21, 2007, the licensee has committed to updating the COLR by adding the 5 rod power restrictions from the September 7, 2007, letter. Based upon these COLR restrictions, the NRC staff finds the fission-product gap fractions used in the AST dose calculations acceptable.

3.1.2.2 Transport

As prescribed in RG 1.183, the PNP FHA is analyzed based on the assumption that 100-percent of the fission products released from the reactor cavity or SFP are released to the environment over a 2 hour period. For the FHA inside containment, the licensee assumed that the equipment hatch is open at the time of the accident and that the release from the containment occurs with no credit taken for containment isolation, no credit for mixing or dilution in the containment atmosphere and no credit for filtration of the released effluent. For the FHA in the FHB, the licensee assumed that a portion of the activity released from the pool water would pass through the FHB filtration system prior to being released to the environment. The licensee analyzed three separated cases to evaluate the effects of partial filtration using 10-percent, 34-percent, and 50-percent of the activity released from pool water processed through the FHB filtration system with elemental and organic iodine filter efficiencies credited at 94-percent. All particulate and aerosol activity is assumed to remain in the reactor cavity and the SFP.

3.1.2.3 CR habitability for the FHA

The licensee evaluated CR habitability for the FHA assuming that the event occurs while the CR ventilation system is operating in the normal mode with 660 cfm of unfiltered airflow into the CR. The licensee assumed that after 20 minutes the CR would be manually isolated and the emergency mode ventilation system would be activated. In the emergency mode, the CR ventilation consists of 1,413.6 cfm of filtered makeup airflow through the emergency intake, and 1,413.6 cfm of filtered recirculation flow. For the FHA CR habitability analysis, the licensee assumed an unfiltered inleakage of 100 cfm after CR isolation. All CR unfiltered inleakage is assessed using the airborne concentrations at the normal CR intake. The licensee credited the CR ventilation filter efficiencies, as applied to both the filtered makeup flow and the recirculation flow, as 99-percent for particulate activity, elemental iodine and organic iodine.

The licensee evaluated the radiological consequences resulting from a postulated FHA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 6 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the FHA meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.3 Main Steam Line Break Accident

The postulated MSLB accident assumes a double-ended break of one main steam line outside the primary containment. This leads to an uncontrolled release of steam from the steam system. The resultant depressurization of the steam system, causes the main steam isolation valves to close and, if the plant is operating at power when the event is initiated, causes the reactor to be scrammed. The radiological consequences of an MSLB outside containment will bound the consequences of a break inside containment. Therefore, only the MSLB outside of containment is considered with regard to the radiological consequences. The affected steam generator (SG), hereafter referred to as the faulted SG, rapidly depressurizes and releases the initial contents of the SG to the environment. The MSLB accident is described in the PNP FSAR Section 14.14.3. RG 1.183, Appendix E, identifies acceptable radiological analysis assumptions for an MSLB.

The licensee calculated the mass of the secondary coolant in the faulted SG based on the maximum hot zero power value. This assumption maximizes the calculated liquid mass release from the faulted SG which maximizes the radiological consequences. The licensee calculated the mass of the secondary coolant in the unaffected SG based on the minimum hot full power value. This assumption maximizes the calculated activity concentration available for release from the unaffected SG. The MSLB evaluation assumes that the PCS mass remains constant throughout the event. The MSLB evaluation credits manual operator action and makeup flow from the auxiliary feedwater system to maintain a constant mass on the secondary side of the intact SG.

3.1.3.1 Source Term

RG 1.183, Appendix E, Regulatory Position 2, states that if no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by TS including the effects of pre-accident and concurrent iodine spiking. Although the licensee's evaluation indicates that no fuel damage is predicted as a result of an MSLB accident, for conservatism the licensee evaluated the MSLB assuming that the event results in 0.5-percent of the fuel experiencing DNB. RG 1.183 also states that the activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum TS values, whichever maximizes the radiological consequences. The licensee determined that the activity released for the MSLB, assuming a 0.5-percent fuel clad failure, exceeds the activity released assuming a release of spiked coolant activity. Therefore the PNP MSLB analysis only considers the more conservative fuel damage case.

The licensee calculated the source term for the MSLB accident by first determining the quantity of activity in the fuel gap by adjusting the total core inventory using the gap fractions for non-LOCA accidents from Table 3 of RG 1.183. The resultant values were then reduced to incorporate the assumption of 2-percent fuel clad failure. The licensee applied a radial peaking factor of 2.04, in accordance with the guidance from RG 1.183, to conservatively bound the source term for the MSLB accident.

In accordance with RG 1.183, Appendix E, Regulatory Position 3, the licensee assumed that all the activity released from the fuel is released instantaneously and homogeneously through the primary coolant. As specified in the RG 1.183, Appendix E, Regulatory Position 4, the licensee assumed that the chemical form of the radioiodine released from the fuel consists of 95-percent CsI, 4.85-percent elemental iodine, and 0.15-percent organic iodine and that the iodine releases from the SGs to the environment consist of 97-percent elemental iodine and 3-percent organic iodine.

Although the release of secondary coolant activity is not specifically addressed in RG 1.183, for the MSLB accident, the licensee evaluated the radiological dose contribution from the release of secondary side activity using the equilibrium secondary side specific activity TS limiting condition for operation (LCO) 3.7.17 of 0.1 micro curie per gram (μCi/gm) Dose Equivalent I-131 (DEI).

3.1.3.2 Transport

The licensee followed the guidance as described in RG 1.183, Appendix E, Regulatory Position 5 in all aspects of the transport analysis for the MSLB. Accordingly, the licensee assumed a primary-to-secondary leak rate in the SGs equal to the leak rate LCO, as specified in the TS, which is 0.3 gpm per SG (LCO 3.4.13). RG 1.183, Appendix E, Regulatory Position 5.2, states that, "The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The alternate repair criteria (ARC) leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specification requirements are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gram per cubic centimeter (gm/cc) (62.4 lbm/ft^3)." The licensee's leak rate testing results are adjusted so that the allowable leakage corresponds to a density of 1.0 gm/cc and accordingly this density was used to convert the volumetric leak rate to a total mass flow rate due to SG tube leakage in the MSLB dose consequence analysis.

RG 1.183, Appendix E, Regulatory Position 5.3, states that, "The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated." In accordance with RG 1.183 and existing licensing bases, the licensee assumed that both primary-to-secondary leakage and releases from the faulted SG continue for 12 hours post MSLB at which time the temperature of the leakage is projected to be less than 100 °C (212 °F) and releases from the faulted SG are terminated. In a letter dated June 15, 2007, the licensee provided additional information indicating that the additional cooldown of 4 hours from shutdown cooling entry conditions to less than 212 °F represents a cooldown rate of less than 25 °F per hour which is considerably less than the TS allowed cooldown rates. Therefore, the licensee asserts and the NRC staff agrees, that the 12 hour time to reach 212 °F is conservative and acceptable for use in the AST MSLB dose consequence analysis. The licensee assumed that the release of radioactivity from the unaffected SG continues for 8 hours until shutdown cooling is in operation and releases from the unaffected SG have been terminated.

In accordance with RG 1.183, the licensee assumed that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix E Regulatory Positions 5.5.1, 5.5.2 and 5.5.3, the licensee assumed that all of the primary-to-secondary leakage into the faulted SG will flash to vapor, and be released to the environment with no mitigation. For the unaffected SG that is used for plant cooldown, the licensee assumed that the primary-to-secondary leakage mixes with the secondary water without flashing.

RG 1.183, Appendix E, Regulatory Position 5.5.4, states that, "The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the PC. A PC for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators." Accordingly, the licensee assumed that the radioactivity in the bulk water of the unaffected SG becomes vapor at a rate that is a function of the steaming rate and the PC. As prescribed by the applicable regulatory guidance, the licensee used a PC of 100 for iodine and other particulate radionuclides.

3.1.3.3 CR ventilation assumptions for the MSLB

The licensee evaluated CR habitability for the MSLB assuming that the event occurs while the CR ventilation system is operating in the normal mode with 660 cfm of unfiltered airflow into the CR. The licensee assumed that after 20 minutes the CR would be manually isolated and the emergency mode ventilation system would be activated. In the emergency mode, the CR ventilation consists of 1413.6 cfm of filtered makeup airflow through the emergency intake, and 1413.6 cfm of filtered recirculation flow. For the MSLB CR habitability analysis, the licensee assumed an unfiltered inleakage of 20 cfm after CR isolation. All CR unfiltered inleakage is assessed using the airborne concentrations at the normal CR intake. The licensee credited the CR ventilation filter efficiencies, as applied to both the filtered makeup flow and the recirculation flow, as 99-percent for particulate activity, elemental iodine and organic iodine.

The licensee evaluated the radiological consequences resulting from the postulated MSLB accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 7 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the MSLB meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.4 Steam Generator Tube Rupture (SGTR)

The licensee evaluated the radiological consequences of an SGTR accident as a part of the full implementation of an AST. The SGTR accident is evaluated based on the assumption of an instantaneous and complete severance of a single SG tube. The postulated break allows primary coolant liquid to leak to the secondary side of the ruptured SG. Integrity of the barrier between the PCS and the main steam system is significant from a radiological release standpoint. The radioactivity from the leaking SG tube mixes with the shell-side water in the affected SG. For the SGTR DBA radiological consequence analysis, a LOOP is assumed to occur shortly after the trip signal, at approximately 700 seconds after the actual SGTR event. Following a reactor trip and turbine trip, the radioactive fluid is released to the environment through the SG atmospheric dump valves (ADVs) or main steam safety valves (MSSVs). Because the LOOP renders the main condenser unavailable, the plant is cooled down by releasing steam to the environment.

3.1.4.1 Source term

Appendix F of RG 1.183 identifies acceptable radiological analysis assumptions for an SGTR accident. If a licensee demonstrates that no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by TSs. Two radioiodine spiking cases are considered. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated SGTR, that has raised the primary coolant iodine concentration to the maximum value permitted by the TS for a spiking condition. For PNP, the maximum iodine concentration allowed by TSs as a result of an iodine spike is 40 μCi/gm DEI.

The second case assumes that the primary system transient associated with the SGTR causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. Initially, the plant is assumed to be operating with the PCS iodine activity at the TS limit for normal operation. For PNP, the PCS TS limit for normal operation is 1 μCi/gm DEI. The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model that assumes that as a result of the accident, iodine is released from the fuel rods to the primary coolant at a rate that is 335 times greater than the iodine equilibrium release rate corresponding to the iodine concentration at the TS limit for normal operation. The iodine release rate at equilibrium is equal to the rate at which iodine is lost due to radioactive decay, PCS purification, and PCS leakage. The iodine release rate is also referred to as the iodine appearance rate. The concurrent iodine spike is assumed to persist for a period of 8 hours.

The licensee's evaluation indicates that no fuel damage is predicted as a result of an SGTR accident. Therefore, consistent with the current licensing analysis basis and regulatory guidance, the licensee performed the SGTR accident analyses for the pre-accident iodine spike case and the concurrent accident iodine spike case. In accordance with regulatory guidance, the licensee assumed that the activity released from the iodine spiking mixes instantaneously and homogeneously throughout the PCS. Additionally, as per RG 1.183 guidance, the licensee assumed that iodine releases from the SGs to the environment consist of 97-percent elemental iodine and 3-percent organic iodine. Although the release of secondary coolant activity is not specifically addressed in RG 1.183, for the SGTR accident, the licensee evaluated the radiological dose contribution from the release of secondary coolant iodine activity at the TS limit of 0.1 µCi/gm DEI.

3.1.4.2 Transport

The licensee followed the guidance as described in RG 1.183, Appendix F, Regulatory Position 5 in all aspects of the transport analysis for the SGTR. Accordingly, the licensee assumed a primary-to-secondary leak rate in the SGs equal to the leak rate LCO, as specified in the TS, which is 0.3 gpm per SG (LCO 3.4.13). RG 1.183, Appendix F, Regulatory Position 5.2, states that, "The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The alternate repair criteria (ARC) leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specification requirements are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3)." The licensee's leak rate testing results are adjusted so that the allowable leakage corresponds to a density of 1.0 gm/cc and accordingly, this density was used to convert the volumetric leak rate to a total mass flow rate due to SG tube leakage in the SGTR dose consequence analysis.

RG 1.183, Appendix F, Regulatory Position 5.3, states that, "The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated." The licensee assumed that the release of radioactivity from both the ruptured SG and the unaffected SG continues for 8 hours, until shutdown cooling is in operation, and steam releases from the steam generators have been terminated. In a letter dated June 15, 2007, the licensee provided additional information regarding the assumption of 8 hours to reach shutdown cooling, explaining that this time period bounds the actual time as indicated by the thermal hydraulic analysis for the SGTR. The thermal hydraulic analysis indicates that a conservative time to reach the shutdown cooling entry conditions would be 23,300 seconds or 6.5 hours. A longer cooldown period results in a greater integrated steam release and is, therefore, conservative with respect to radiological consequences.

The licensee evaluated the dose consequences from discharges of steam from the intact SG for a period of 8 hours, until the primary system has cooled sufficiently to allow an alignment to the shutdown cooling system. At this point in the accident sequence, steaming is no longer required for cool down and releases from the intact SG are terminated.

The licensee assumed that the source term resulting from the radionuclides in the primary system coolant, including the contribution from iodine spiking, is transported to the ruptured SG by the break flow. A portion of the break flow is assumed to flash to steam because of the higher enthalpy in the RCS relative to the secondary system. The flashed portion of the break flow will ascend through bulk water of the SG, enter the steam space of the affected generator, and be immediately available for release to the environment. The licensee credited scrubbing of the particulate activity in the break flow in the ruptured SG. The licensee used the methodologies described in NUREG-0409 to determine the amount of scrubbing credit applied to the flashed portion of the break flow.

The licensee determined the amount of liquid available for scrubbing by using the integrated ADV and ruptured SG tube flow rates. Due to the presence of a two phase mixture above the tube bundle, the licensee did not credit scrubbing until after the turbine valves were completely closed at 707.1 seconds into the event. The licensee assumed that the initial water level above the tubes after closure of the turbine valves was zero. The time dependent water level above the top of the tubes was then determined by deducting the ADV release from the liquid break flow. The volume of water added was determined by converting the added mass to a volume based on the saturation pressure in the steam generator. The added volume was then divided by the SG cross sectional area to obtain the height of water. The licensee minimized the scrubbing credit by maximizing the calculation of the SG cross sectional area. To accomplish this the licensee used the outside diameter of the SG and ignored the space occupied by the SG internals.

During the first 707.1 seconds of the event, prior to the LOOP, the licensee assumed that all steam flows are routed to the condenser. The licensee applied a DF of 100 to the releases of particulates and elemental iodine from the condenser. The use of a DF of 100 as applied to the flashed flow for the first 707.1 seconds of the event, while the steam flows are routed to the condenser, is a typical and conservative approach that is acceptable to the NRC staff.

The licensee modeled steam releases from the affected SG from until the ruptured SG is completely depressurized. For the time period from 707.1 seconds to 1800 seconds the licensee modeled the steam release via the most limiting MSSV. After 1800 seconds, the licensee modeled the steam release via the most limiting ADV. Depressurization of the SG is necessary to allow for the implementation of shutdown cooling system cooling.

The iodine and other non-noble gas isotopes in the non-flashed portion of the break flow are assumed to mix uniformly with the SG liquid mass and be released to the environment in direct proportion to the steaming rate and in inverse proportion to the applicable PC.

The licensee has determined that for the SGTR accident, both SGs effectively maintain tube coverage. In accordance with RG 1.183, Appendix E, Regulatory Position 5.5.1, the licensee assumed that for the ruptured SG, and the unaffected SG used for plant cooldown, the primary-to-secondary leakage mixes with the secondary water without flashing due to the total submergence of the SG tubes. The iodine and other non-noble gas isotopes in the primary-to-secondary leakage flow are assumed to mix uniformly with the SG liquid mass and be released to the environment in direct proportion to the steaming rate and in inverse proportion to the applicable PC. In accordance with the guidance from RG 1.183, the licensee's evaluation of the releases from the steaming of the liquid mass in the SGs credits a PC of 100 for all non-noble gas isotopes. Following the applicable regulatory guidance, the licensee assumed that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation.

3.1.4.3 CR ventilation assumptions for the SGTR

As explained in the licensee's SGTR radiological analysis, the unfiltered makeup flow covers three time periods. A LOOP is assumed to occur coincident with the SGTR. According to Table 14.15-3 of the FSAR, offsite power is lost at 708.2 seconds. The unfiltered flow rate during the period of lost offsite power consists of the base infiltration rate of 384.2 cfm. The CR normal ventilation rate of 660 cfm is assumed to be restored at 798.2 seconds which is the summation of the 708.2 seconds plus an additional 90 seconds assumed for diesel generator startup and the restart of normal CR ventilation. At 20 minutes post-SGTR, the CR is assumed to be isolated by manual activation of the CR emergency filtration system. Since the time period without offsite power is only 90 seconds, and since the infiltration flow rate of 384.2 cfm during this period is lower and would reduce the quantity of radionuclides entering the CR, the licensee conservatively assumed that the unfiltered makeup flow remains constant at 660 cfm until the CR enters the emergency mode.

In the emergency mode, the CR ventilation consists of 1,413.6 cfm of filtered makeup airflow through the emergency intake, and 1,413.6 cfm of filtered recirculation flow. For the SGTR CR habitability analysis, the licensee assumed an unfiltered inleakage of 100 cfm after CR isolation. All CR unfiltered inleakage is assessed using the airborne concentrations at the normal CR intake.

The licensee credited the CR ventilation filter efficiencies, as applied to both the filtered makeup flow and the recirculation flow, as 99-percent for particulate activity, elemental iodine and organic iodine.

The licensee evaluated the radiological consequences resulting from the postulated SGTR accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 8 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the SGTR accident meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.5 Small Line Break Outside Containment (SLBOC)

The SLBOC event postulates the break of a 2-inch RCS letdown line in the auxiliary building outside of the containment. The letdown line is the largest piping that carries RCS fluid outside containment. A rupture of the letdown line provides a release path for the primary coolant to the environment through the auxiliary building ventilation system that exhausts to the plant stack. The licensee assumed a break flow rate of 160 gpm at a temperature of 130 °F and a pressure of 35 pounds per square inch absolute (psia). The licensee assumed that 60 minutes would be required to identify and isolate the break flow. The licensee assumed that the reactor does not trip for the SLBOC evaluation.

Neither RG 1.183, nor SRP 15.0.1, include the SLBOC event as a DBA. Therefore, the licensee followed the methods employed in SRP, Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," with appropriate modifications to maintain consistency with the assumptions in RG 1.183. The licensee used the most restrictive acceptance criteria from SRP 15.0.1, Table 1 and RG 1.183, Table 6, for application in the SLBOC event. The approach taken by the licensee, to evaluate the SLBOC accident, using the applicable guidance from SRP, Section 15.6.2 and using the most restrictive acceptance criteria for any of the DBAs considered, is conservative and, therefore, acceptable to the NRC staff.

3.1.5.1 Source term

The licensee's analysis assumed that no fuel failure results from the letdown line break, which is consistent with the current licensing basis in the PNP FSAR Section 14.23. In accordance with the assumptions of the current analysis of record for this event, the licensee did not assume that the SLBOC results in a reactor trip. Initially, the radioactivity in the RCS is assumed to be at the equilibrium iodine TS limit of 1.0 μCi/gm DEI. The consideration of the equilibrium iodine TS limit is consistent with the review procedure provided in SRP, Section 15.6.2. Following the guidance in SRP, Section 15.6.2, the licensee assumed that the primary system transient associated with the SLBOC causes a concurrent iodine spike in the primary system. The licensee estimated the increase in primary coolant iodine concentration for the concurrent iodine spike using a spiking model that assumes that as a result of the accident, iodine is released from the fuel rods to the primary coolant at a rate that is 500 times greater than the iodine equilibrium release rate corresponding to the iodine concentration at the TS limit for normal operation. The iodine release rate at equilibrium is equal to the rate at which iodine is lost due to radioactive decay, PCS purification, and PCS leakage. The concurrent iodine spike is assumed to persist for a period of 8 hours.

3.1.5.2 Transport

The licensee determined that for a break flow rate of 160 gpm at 130 °F and 35 psia, the resulting mass flow rate would be 1314.78 lbm/min. The licensee assumed that the break flow would persist for a period of 1 hour. The licensee maintained the assumptions of a break flow rate of 160 gpm and an isolation time of 1 hour that were used in the technical review of process lines carrying primary coolant outside of containment performed for the systematic evaluation program. The licensee calculated, based on the thermodynamic conditions of the break flow, that there would be no flashing of the released fluid. However, the licensee used the guidance provided in RG 1.183, Appendix A, Regulatory Positions 5.4 and 5.5, pertaining to ESF leakage during a LOCA, and conservatively assumed that 10-percent of the total iodine in the leaked fluid would be released to the auxiliary building atmosphere. In accordance with RG 1.183, the licensee assumed that the chemical form of the released iodine is 97-percent elemental and 3-percent organic. The licensee assumed that all of the noble gas activity in the break flow is released to the auxiliary building atmosphere. The licensee assumed that the remaining particulate activity would remain in the liquid phase.

3.1.5.3 CR ventilation assumptions for the SLBOC

The licensee evaluated CR habitability for the SLBOC assuming that the event occurs while the CR ventilation system is operating in the normal mode with 660 cfm of unfiltered airflow into the

CR. The licensee assumed that after 20 minutes the CR would be manually isolated and the emergency mode ventilation system would be activated. In the emergency mode, the CR ventilation consists of 1,413.6 cfm of filtered makeup airflow through the emergency intake, and 1,413.6 cfm of filtered recirculation flow. For the SLBOC CR habitability analysis, the licensee assumed an unfiltered inleakage of 100 cfm after CR isolation. All CR unfiltered inleakage is assessed using the airborne concentrations at the normal CR intake. The licensee credited the CR ventilation filter efficiencies, as applied to both the filtered makeup flow and the recirculation flow, as 99-percent for particulate activity, elemental iodine and organic iodine.

The licensee evaluated the radiological consequences resulting from the postulated SLBOC accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and meet the most restrictive accident dose criteria specified in SRP Section 15.0.1, Table 1 and RG 1.183, Table 6. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 9 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the SLBOC accident meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.6 Control Rod Ejection Accident (CREA)

Section 14.16 of the PNP FSAR describes the CREA as the mechanical failure of a control rod mechanical pressure housing such that the coolant system pressure ejects a control rod blade assembly and drive shaft to a fully withdrawn position. The consequences of this mechanical failure are a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Following the applicable guidance, the licensee evaluated two separate release scenarios for the CREA. In the first case, the CREA is assumed to induce a LOCA resulting in a release of fission products into the containment atmosphere and a subsequent release to the environment from the containment leakage pathway.

For the second case, the radiological consequences from a CREA are evaluated assuming that the PCS boundary remains intact and that fission products are released to the environment from the secondary system. In this case fission products from the damaged fuel are assumed to be released to the primary coolant and transported to the secondary system through primary to secondary leakage in the SGs. The CREA is analyzed with the assumption of a concurrent LOOP which causes the MSSVs to lift and release steam from the secondary system to the environment.

3.1.6.1 Source term

The source term for the CREA is assumed to result in fuel damage consisting of localized damage to fuel cladding with a limited amount of fuel melt occurring in the damaged rods. The source term for the CREA is described in RG 1.183, Appendix H, Regulatory Position 1, which states that, "Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10-percent of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100-percent of the noble gases and 25-percent of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100-percent of the noble gases and 50-percent of the iodines in that fraction are released to the reactor coolant."

The licensee used the AST source term inventory from the LOCA, with an average burnup of 39,300 MWD/MTU and a core power level of 2,703 MWt, as a basis for developing the source term for use in the CREA. The licensee assumed that as a result of the CREA, 14.7-percent of the fuel experiences DNB and 0.5-percent of the fuel experiences fuel centerline melt (FCM). These values bound the values calculated in the fuel failure analysis described in PNP FSAR Section 14.16.2.3. The FSAR analysis predicts that 7.2-percent of the fuel rods in the core are expected to fail due to DNB considerations and none of the fuel rods in the core are predicted to fail due to FCM considerations, for the most limiting CREA case. The licensee applied a radial peaking factor of 2.04 in the development of the CREA source term.

Consistent with the guidance provided in RG 1.183, Appendix H, the licensee assumed that 10-percent of the core inventory of noble gases and iodine reside in the fuel gap. The licensee assumed that 12-percent of the core inventory of alkali metals, which includes cesium and rubidium, reside in the fuel gap, as specified in Table 3 of RG 1.183. The licensee assumed that for the 14.7-percent of the fuel that experiences DNB, all of the gap activity contained in the affected fuel will be available for release in both the CREA induced LOCA scenario and the secondary side release scenario.

In accordance with RG 1.183, for the 0.5-percent of fuel experiencing FCM, the licensee assumed that 100-percent of the noble gases, 25-percent of the iodines, and 12-percent of the alkali metals in the affected fuel will be available for release from containment in the CREA induced LOCA scenario. For the 0.5-percent of fuel experiencing FCM, the licensee assumed that 100-percent of the noble gases, 50-percent of the iodines, and 12-percent of the of the alkali metals in the affected fuel will be available for release from the RCS for the secondary side CREA release scenario. In accordance with RG 1.183, Appendix H, Regulatory Position 3, 100-percent of the released activity is assumed to be released instantaneously and mixed homogeneously throughout the containment atmosphere for the CREA induced LOCA; and 100-percent of the released activity is assumed to be released instantaneously and completely dissolved in the primary coolant and available for release to the secondary system in the CREA secondary side release scenario.

In accordance with RG 1.183, Appendix H, Regulatory Position 4, the licensee assumed that the chemical form of radioiodine released to the containment atmosphere consists of 95-percent CsI, 4.85-percent elemental iodine, and 0.15-percent organic iodine. The licensee credits effective controls to limit the pH in the containment sump to 7.0 or higher. In the application, the licensee committed to implement a buffer program to maintain a pH of 7.0 - 8.0 post-LOCA with recirculation, during the 2007 fall refueling outage at PNP.

In accordance with RG 1.183, Appendix H, Regulatory Position 5, the licensee assumed that the chemical form of radioiodine released from the SGs to the environment atmosphere consists of 97-percent elemental iodine, and 3-percent organic iodine.

Although the release of secondary coolant activity is not addressed in RG 1.183, for the CREA, the licensee evaluated the radiological dose contribution from the release of secondary coolant iodine activity at the TS limit of 0.1 μCi/gm DEI.

3.1.6.2 Transport from containment

For the CREA induced LOCA case, the licensee credited natural deposition in the containment. The licensee evaluated the natural deposition in the containment using an aerosol removal coefficient of 0.1 per hour and an elemental iodine removal coefficient of 1.3 per hour. The licensee did not credit the use of containment sprays for fission product removal in the CREA. In accordance with RG 1.183, the licensee assumed that the containment would leak at the proposed TS maximum rate of 0.10 weight percent per day for the first 24 hours of the accident, and at 0.05 weight percent per day for the remainder of the event.

3.1.6.3 Transport from secondary system

In accordance with RG 1.183, Appendix H, Regulatory Position 7, the licensee evaluated the transport of activity from the PCS to the SGs secondary side, assuming the maximum TS primary-to-secondary leak rate of 0.3 gpm. The licensee assumed that this leak rate persists for a period of 8 hours until shutdown cooling is in operation and releases from the SGs have been terminated. The licensee's leak rate testing results are adjusted so that the allowable leakage corresponds to a density of 1.0 gm/cc and accordingly this density was used to convert the volumetric leak rate to a total mass flow rate due to SG tube leakage in the CREA dose consequence analysis.

In accordance with RG 1.183, the licensee assumed that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix E Regulatory Positions 5.5.1, 5.5.2 and 5.5.3, the licensee assumed that all of the primary-to-secondary leakage in both SGs mixes with the secondary water without flashing.

In accordance with RG 1.183, Appendix E, Regulatory Position 5.5.4, the licensee assumed that the radioactivity in the bulk water of both SGs becomes vapor at a rate that is a function of the steaming rate and the PC. As per regulatory guidance, the licensee used a PC of 100 for iodine and other particulate radionuclides.

3.1.6.4 CR ventilation assumptions for the CREA

As explained in the licensee's CREA radiological analysis, the CR ventilation system cycles through three time periods. A LOOP is assumed to occur coincident with the CREA. The unfiltered flow rate during the initial period of lost offsite power consists of the base unfiltered infiltration rate of 384.2 cfm. The CR normal unfiltered ventilation flow rate of 660 cfm is restored at 90 seconds, which is the time allotted for diesel generator startup and normal ventilation restart. The licensee assumed that 20 minutes after the CREA, the CR would be manually isolated and the emergency mode ventilation system would be activated. In the emergency mode, the CR ventilation consists of 1413.6 cfm of filtered makeup airflow through the emergency intake, and 1413.6 cfm of filtered recirculation flow. For the CREA CR habitability analysis, the licensee assumed an unfiltered inleakage of 10 cfm after CR isolation. All CR unfiltered inleakage is assessed using the airborne concentrations at the normal CR intake. The licensee credited the emergency mode CR ventilation filter efficiencies, applied to both the filtered makeup flow and the recirculation flow, as 99-percent for all particulate activity, elemental iodine and organic iodine.

The licensee evaluated the radiological consequences resulting from the postulated CREA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 10 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the CRE were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.7 Spent Fuel Cask Drop (SFCD) Accident

Postulated cask drop accidents at PNP are described in FSAR Section 14.11, which states that, "In 2003, Facility Change FC-976 modified the main hoist of the Fuel Building Crane to increase the capacity to 110-tons, and to meet single failure criteria in accordance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." Postulated load drops from the SFP area cranes are analyzed, unless the crane and lifting devices are designed and specified to be single failure proof in accordance with NUREG-0612 and NUREG-0554. Since the main hoist has been upgraded to meet single-failure-proof criteria, analyses of postulated load drops from the main hoist are no longer required."

"Although the main hoist of the spent fuel crane is designed and operated in accordance with single-failure-proof criteria for cask handling activities, there may be situations in which lifting devices used with the main hook do not meet these requirements or single-failure-proof features of the main hoist may be disabled. In these situations, the crane no longer meets single failure proof requirements, and load drops are postulated. Therefore, this section contains an outline of the methodology and evaluations used to document the consequences of postulated fuel transfer cask drop accidents in the fuel handling area of the Palisades plant."

"Since the 15-ton auxiliary hoist of the SFP crane is not single failure proof, postulated load drops from the auxiliary hoist have been evaluated in accordance with NUREG-0612. Heavy Loads handled with the auxiliary hoist are limited to designated safe load paths."

The licensee has determined that the evaluation of the radiological consequences for a postulated load drop of a loaded Multi-Assembly Sealed Basket Transfer Cask (MTC) from the main hoist, as described herein, bounds the radiological consequences from postulated load drops from the auxiliary hoist.

The licensee analyzed three cask drop scenarios using various release pathways and decay times to bound the radiological dose consequence analysis. For each scenario, the cask is assumed to impact the stored spent fuel assemblies and result in the release of fission products contained within the fuel gap of the stored fuel. No damage is postulated for the fuel being transferred in the MTC. The SFCD is not addressed in RG 1.183 or in SRP, Section 15.0.1.

The licensee used the methodology described in RG 1.183, Appendix B, which outlines the requirements for performing a radiological analysis of an FHA, to perform the dose consequence analyses for the SFCD accident. The licensee also applied the dose acceptance criteria for an FHA as described in SRP, Section 15.0.1, Table 1 and RG 1.183, Table 6. It should be noted that in performing the radiological consequences of a SFCD in the SFP, the licensee did not take credit for the use of an impact limiting pad in the SFP.

SRP Section 15.7.4, "Radiological Consequences of a Fuel Handling Accident," Revision 1, July 1981, covers the review of the radiological consequences of a postulated FHA and states that, "Such accidents include the dropping of a single fuel assembly and handling tool or of a heavy object onto other spent fuel assemblies." Since the PNP SFCD accident does not postulate the damage of the fuel being transferred in the MTC, the accident should be evaluated as the drop of a heavy object onto other spent fuel assemblies. Therefore, the NRC staff finds that the licensee's approach to the evaluation of the SFCD, using the same guidance and dose acceptance criteria as is used for a FHA, is acceptable.

3.1.7.1 Source term

For the SFCD accident, the licensee has defined the event as a cask drop onto stored spent fuel that results in the damage to all of the fuel pins in 73 fuel assemblies. The licensee evaluated the per assembly source term for the SFCD using the same conservative approach used for the FHA with the exception of the decay times considered. Following the same approach taken for the FHA, the licensee adjusted the activities for I-131 and Kr-85 to account for different gap release fractions for these isotopes as specified by Table 3 of RG 1.183. To determine the activity contained in the damaged assemblies, the licensee multiplied the maximized per assembly decayed activity by 73. This approach is conservative in that the per assembly activity was determined by considering the maximum activity on a per nuclide basis as described in the FHA. The licensee's evaluation of the source term for the SFCD, which maximizes the activity for each isotope in each of the 73 damaged assemblies, is a conservative approach and is, therefore, acceptable to the NRC staff.

Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the SFP depending on their physical and chemical form. Following the guidance in RG 1.183, Appendix B, Regulatory Position 1.3, the licensee made the following assumptions: the chemical form of radioiodine released from the fuel to the SFP consists of 95-percent CsI, 4.85-percent elemental iodine, and 0.15-percent organic iodine; the CsI released from the fuel is assumed to completely dissociate in the pool water; because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This results in a final iodine distribution of 99.85-percent elemental iodine and 0.15-percent organic iodine. The licensee assumed that the release to the pool water and the chemical redistribution of the iodine species occurs instantaneously.

3.1.7.2 Transport

The licensee considered two possible release paths for the activity that escapes the fuel pool in the FHB; an unfiltered release that travels to the containment and exits via the containment equipment hatch, and a filtered release via the FHB ventilation system which exhausts via the plant stack. For the filtered release path via the FHB ventilation system, the licensee considered two separate cases with different percentages of effluent filtration. Therefore, the licensee analyzed three cases for the transport of released activity from the damaged spent fuel to the environment:

- Case 1 with 90-percent of the release via the FHB filtration system, 30 days of decay, and the control room initially aligned in emergency recirculation mode.

- Case 2 with 82.5-percent of the release via the FHB filtration system, 30 days of decay, and the control room initially aligned in emergency recirculation mode.

- Case 3 with 0-percent of the release via the FHB filtration system, 90 days of decay, and no isolation of the control room.

The licensee did not credit mixing of the release from the fuel pool with the FHB atmosphere or the containment atmosphere. The licensee asserts that at all times, a minimum of 23.4 feet of water is maintained above the fuel stored in the SFP thereby allowing an overall iodine DF of 200 as per the guidance in RG 1.183, Appendix B, Regulatory Position 2. In accordance with RG 1.183, Appendix B, Regulatory Position 4.1, the licensee assumed the release occurs over a 2-hour period.

3.1.7.3 CR habitability for the SFCD

In a letter dated June 15, 2007, the licensee provided additional information regarding the modeling of the SFCD in relation to the assumptions for the evaluation of CR habitability. For cases 1 and 2, the licensee evaluated CR habitability based on the condition that the CR is initially aligned in the emergency mode. In the emergency mode, the CR ventilation consists of 1,413.6 cfm of filtered makeup airflow through the emergency intake, and 1,413.6 cfm of filtered recirculation flow. For cases 1 and 2 of the SFCD CR habitability analysis, the licensee assumed an unfiltered inleakage of 100 cfm for the duration of the accident. All CR unfiltered inleakage is assessed using the airborne concentrations at the normal CR intake. The licensee credited the CR ventilation filter efficiencies, as applied to both the filtered makeup flow and the recirculation flow, as 99-percent for particulate activity, elemental iodine and organic iodine.

For case 3 the licensee evaluated CR habitability for the SFCD assuming that the event occurs while the CR ventilation system is operating in the normal mode with 660 cfm of unfiltered airflow into the CR. The licensee assumed that the CR ventilation system would be maintained in the normal mode for the duration of the accident.

The licensee evaluated the radiological consequences resulting from the postulated SFCD and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and the accident specific dose criteria specified in SRP, Section 15.0.1 for the FHA. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 11 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the SFCD were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.2 Atmospheric Dispersion

The licensee calculated new atmospheric dispersion factors, referred to as χ/Q values, for use in evaluating the radiological consequences of DBAs for the CR and offsite exposures. The ARCON96 atmospheric dispersion model as described in NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes," was used to calculate onsite χ/Q values at multiple release-receptor pairs for each of the DBAs analyzed. The PAVAN code as described in NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," was used to calculate new χ/Q values for use in evaluating the radiological consequences of DBAs at the EAB and outer boundary of the LPZ. The resulting suite of CR and offsite χ/Q values denote a change from the previous set of χ/Q values currently presented in the PNP UFSAR.

3.2.1 Meteorological Data

The licensee collected five consecutive years of onsite meteorological data, from 1999 through 2003, that were used to generate new LOCA, MSLB, SGTR, SLBOC, CREA, FHA, and SFCD CR and offsite χ/Q values for this AST LAR. Hourly meteorological data, including wind speed, wind direction and atmospheric stability class, were submitted, which enabled the NRC staff to perform independent, confirmatory analyses. These hourly meteorological data served as input to ARCON96, in addition to being used to generate joint frequency distributions (JFDs) of wind speed and wind direction with respect to atmospheric stability class for input to PAVAN.

The licensee asserts that the onsite meteorological measurement program complies with the guidelines set forth in RG 1.23, "Onsite Meteorological Programs," 1972. The atmospheric dispersion analyses were performed using wind measurements taken at 10.1 meters (m) and 57.8 m above ground level (AGL). Atmospheric stability, parameterized using the Pasquill-Gifford stability categories, was derived from the difference in air temperature, referred to as delta-temperature, measured between the 57.8 m and 10.1 m levels. Analysis by the NRC staff confirmed that the meteorological data submitted resulted in a combined 5-year data recovery rate of approximately 99-percent, well above the 90-percent capture rate specified by RG 1.23. In a letter dated June 15, 2007, the licensee provided additional information regarding the validity of the meteorological data submitted to the NRC. The licensee confirmed that the occurrence of a wind direction-wind speed pair each reporting a zero value was indeed valid. The licensee stated that a wind speed recorded as 0 m per second (m/s) will result in an undefined wind direction, which was reported as 0°. The ARCON96 code characterizes a wind direction equal to 0 as invalid. However, according to NUREG/CR-6331, Revision 1, and from the ARCON96 source code, specifically subroutine XOQCALC5, for calm conditions with wind speeds less than 0.5 m/s as per RG 1.194, the wind direction is not relevant and a low wind speed correction is applied, the χ/Q is calculated, and the count of calm conditions is updated.

The NRC staff performed a quality review of the 1999 through 2003 hourly meteorological data. The 1999 through 2003 wind speed, wind direction, and atmospheric stability frequency distributions were consistent from year to year. The results exhibited characteristic behavior across a multitude of varying temporal and spatial scales. For example, the onset and duration of observed stability regimes were generally consistent with expected meteorological conditions, including stable and neutral conditions usually occurring during the nighttime hours and unstable and neutral conditions usually occurring during the daytime hours.

The NRC staff analyzed and compared the onsite December 1977 through November 1978 wind speed, wind direction, and stability data summaries presented in Chapter 2, Section 2.5, Revision 25 of the PNP FSAR, with the onsite data collected from January 1999 through December 2003. The analysis illustrated good overall agreement between the two data sets, even though the initial data set contained only 1 year of measurements. For example, both the December 1977 through November 1978 and the January 1999 through December 2003 data sets were dominated by similar flow regimes, including wind direction frequency, at both the 10.1 m and 57.8 m levels. In general, the stability frequency distribution was found to be in good agreement, with the only deviations being an approximate 10-percent increase in the A stability class, from the December 1977 through November 1978 data set to the January 1999 through December 2003 data set, and an approximate 7-percent decrease in the E stability class, from the December 1977 through November 1978 data set to the January 1999 through December 2003 data set.

For the reasons previously stated, the NRC staff concludes that the 1999 through 2003 onsite meteorological database submitted to the NRC by the licensee is acceptable for deriving atmospheric dispersion estimates for use in the DBA dose assessments performed in support the PNP proposed LAR for AST implementation, dated September 25, 2006.

3.2.1.1 CR atmospheric dispersion factors

PNP's CR HVAC system operates in three different modes: normal, emergency, and purge. During the CR normal mode of operation, the CRE is slightly pressurized with respect to the surroundings, allowing unfiltered outside air to be continuously introduced into the CRE from two possible normal outside air intakes. In emergency or recirculation mode, outside air enters the CR through the emergency ventilation air intake. The emergency mode of operation can be initiated by a containment high radiation signal, a safety injection signal, or manually by the plant operator. In this particular mode of operation, normal outside air intakes are closed and the CR is pressurized at a greater rate than the surroundings in order to maintain a positive pressure difference. Fresh and recirculated air is sent through high efficiency particulate filters and charcoal filters to maintain an adequate CR environment. During purge mode, air is brought into the CR at a significantly higher rate than any of the other modes to prevent recirculation of air within the CR.

The CR dose calculation model includes a recirculation filter model, filtered air intake, unfiltered air inleakage, and an exhaust pathway. It should be noted that <u>two</u> different values of unfiltered inleakage are being utilized in the CR dose model depending on how limiting the event tends to be. The first assumption, 10 cfm unfiltered inleakage, applies to the more limiting events that include the LOCA, MSLB, and CREA, where radiological dose margin is lower. On the other hand, the second assumption of 100 cfm unfiltered inleakage, applies to the less limiting incidents that include the SGTR, SLBOC, FHA, and SFCD, where radiological dose margins are greater. There is concern over the use of 10 cfm as the unfiltered inleakage term in the CR dose model because of the already small margin with regards to the CR dose calculated for the MSLB of 4.98 rem and the acceptance criteria of 5.0 rem as specified in 10 CFR 50.67. PNP's LAR describes the results of a tracer gas experiment that demonstrated an unfiltered inleakage rate less than what was assumed in the FSAR analyses. According to the LAR, the use of 10 cfm as a design basis value will be established to be above the unfiltered inleakage value determined through modification, testing and analysis consistent with the resolution of issues identified in NEI [Nuclear Energy institute] 99-03 and Generic Letter 2003-01.

The licensee used guidance provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," to generate new control room atmospheric dispersion factors. The ARCON96 computer code was used to calculate new χ/Q values for multiple release-receptor combinations, to evaluate the radiological consequences of the DBAs on the CR as described in Section 3.2, "Atmospheric Dispersion Factors," above. RG 1.194 suggests that ARCON96 is an acceptable methodology for evaluating CR χ/Q values for use in DBA radiological analyses. The release points were as follows: (1) closest containment point, (2) SIRWT vent, (3) stack vent, (4) closest atmospheric dump valve (ADV), (5) closest safety relief valve (SRV), (6) containment equipment door, (7) turbine building NE roof exhaust, (8) turbine building NW roof exhaust and (9) V-22A/B east and west unit release locations. The receptor points evaluated include the CR normal intake A, CR normal intake B, and the CR emergency intake. For all of the DBAs evaluated, the more conservative normal intake, A or B, χ/Q values were used to model the CR dose prior to the isolation of the CR. In addition, the CR emergency intake χ/Q values were used to model the CR dose following the isolation of the CR. Details, including the release height, receptor height, distance, and direction used in the licensee's assessments of CR post-accident atmospheric dispersion conditions for the resultant χ/Q values were initially summarized in Table 1.8.1-1 "Release - Receptor Combination Parameters for Analysis Events," in Enclosure 4 of NAI Report No. NAI-1149-027 AST Licensing Technical Report for Palisades, Revision 1 (ADAMS Accession No. ML062830385) of the PNP proposed LAR for AST implementation dated September 25, 2006. In order to perform a confirmatory analysis of the χ/Q values, derived using ARCON96, submitted in the PNP proposed LAR for AST implementation dated September 25, 2006, NRC staff requested supplemental plant drawings which were provided by the licensee in a letter dated June 15, 2007. As part of the effort to provide clarifying information on the location of postulated release/receptor pairs, the licensee determined that information for some of the release/receptor pairs was not accurate and provided revised information by letter dated September 7, 2007. In addition, the licensee also identified the V-22A/B vents as two additional release locations for the MSLB and provided relevant release/receptor information and revised χ/Q values by letter dated September 20, 2007.

The onsite meteorological data including wind speed, direction and delta temperature that were collected during the calendar years of 1999 through 2003 at the 10.1 m and 57.8 m levels were used as input into the ARCON96 code. The licensee acknowledged that all the default values to the ARCON96 code were used with the exception of 0.2 for the surface roughness length and 4.3 for the averaging sector width constant from Table A-2 of RG 1.194. In summary, the licensee asserts that the following conservatism assumptions were used in determining the CR χ/Q values: (1) ARCON96 was used to calculate the χ/Q values, (2) all accidents were evaluated as ground level releases since none of the release points are 2.5 times taller than the closest solid structure; therefore, vertical velocity, stack flow, and stack radius were set equal to zero in ARCON96, (3) diffuse area was not assumed for any release pathway, (4) plume rise was not modeled, and (5) building wake was only credited for accidents involving the containment building or the SIRWT releases and conservative building areas were used in these evaluations.

In a letter dated June 15, 2007, the licensee provided additional information acknowledging a correction to results presented in Table 1.8.1-3 of NAI Report No. NAI-1149-027 Revision 1, for the Small Line Break Outside Containment. The release-receptor pair, "Prior to CR Isolation" and "Following CR Isolation" is incorrectly labeled as "F" and "E", respectively. The release-receptor pair describing the Small Line Break Outside Containment event, "Prior to CR

Isolation" and "Following CR Isolation," should be reversed and labeled "E" and "F", respectively. The licensee assured NRC staff that the correct χ/Q values were used in the calculations and that this was simply a typographical error.

The NRC staff evaluated the applicability of the ARCON96 model. With the exception of the SIRWT Vent Normal Control Room Intake B release-receptor combination, the NRC staff concluded that there is no other unusual siting, building arrangements, release characterizations, release-receptor combinations, meteorological regimes, or topography that precludes the use of the ARCON96 model for the Palisades site. The PNP has a unique CR design in which the SIRWT is located directly above the CR and in close proximity to the CR normal intakes. The licensee evaluated the direct dose to the CR from back-leakage of contaminated sump water to the SIRWT during post-accident sump recirculation. In addition, the licensee evaluated the inhalation and immersion doses into the CRE resulting from post-LOCA releases from the SIRWT vent. The NRC staff expressed a concern regarding the SIRWT Vent-Normal CR Intake B release-receptor combination. The evaluation was based on a distance of 7.7-m from the SIRWT Vent to the Normal Control Room Intake B. According to the RG 1.194, Regulatory Position C.3.4, if the distance to receptor is less than about 10 m, ARCON96 should not be used to assess relative concentrations. In a letter dated June 15, 2007, the licensee acknowledged that the release-receptor combination involving the SIRWT Vent and Normal Control Room Intake B is less than 10 m. The licensee asserts that the SIRWT Vent to Normal Control Room Intake B, release-receptor combination χ/Q values as calculated using ARCON96, are more conservative than the previously accepted FSAR χ/Q values. The licensee also included results for the release-receptor combination involving the SIRWT Vent and Normal CR Intake B from an independent wind tunnel test, which illustrates that the χ/Q values proposed for AST implementation are more conservative than those χ/Q values calculated from the wind tunnel test data. In addition, any uncertainty in χ/Q values calculated for the AST implementation resulting from utilizing ARCON96 with a release-receptor distance of less than 10 m should not substantially impact the dose assessment because: (1) the source term has not fully developed in 90 seconds, (2) after the beginning of the event and the start of the diesel generators the CR is isolated, and (3) unfiltered inleakage through the normal intakes is assumed to be no more than 10 cfm for a LOCA event.

The NRC staff performed a confirmatory analyses with regards to the licensee's assessments of the CR post-accident dispersion conditions generated using the licensee's meteorological data as input to ARCON96. The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with the site configuration drawings, other information provided by the licensee, and NRC staff practice. The χ/Q values generated by the licensee are presented in Table 2 below and in Table 1.8.1-2, NAI Report No. NAI-1149-027, Revision 1, of the September 25, 2006, letter, with the addition of χ/Q values for the V-22A/B release locations as provided in the September 20, 2007, letter. In the September 20, 2007, letter, the licensee also provided revised χ/Q values using the revised release/receptor information provided in the September 7, 2007, letter, but these revised χ/Q values were not used in the dose assessment of this LAR. However, the revised χ/Q values were used to substantiate the acceptability of the initial χ/Q values that were used in the dose assessment. The NRC staff calculated similar χ/Q values when performing its confirmatory analyses and, therefore, concludes that the χ/Q values calculated by the licensee for DBA releases to the PNP CR are acceptable for use in the DBA CR dose assessments performed in support of the PNP proposed LAR for AST implementation.

3.2.1.2 EAB and LPZ atmospheric dispersion factors

The licensee used the PAVAN computer code to generate χ/Q values at the EAB and LPZ using a JFD of wind direction and wind speed with respect to atmospheric stability class. The licensee derived EAB and LPZ χ/Q values at downwind distances of 677 m and 4820 m, respectively. The PAVAN computer code implements the guidance provided in RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.”

A JFD was generated by the licensee, comprised of wind speed and wind direction with respect to atmospheric stability class data derived from the meteorological data for the years 1999 through 2003. This JFD was used as input into the PAVAN code. Wind speed and wind direction measurements collected from the meteorological tower at 10.1 m AGL were used. In addition, stability class was derived from the difference in air temperature measured between 57.8 m AGL and 10.1 m AGL. The NRC staff also generated a JFD from the hourly meteorological database submitted by the licensee and obtained similar results. The licensee’s wind speed categories deviated from the finer wind speed category breakdown, especially at low wind speeds, suggested in the NRC Regulatory Issue Summary (RIS) 2006-04, “Experience with Implementation of Alternative Source Terms,” dated March 7, 2006 (ADAMS Accession No. ML053460347). However, NRC staff’s confirmatory calculations using the suggested finer wind speed category breakdown in RIS 2006-04 resulted in χ/Q estimates that were similar to those calculated by the licensee.

Since the highest possible release height is less than 2.5 times higher than the adjacent containment building, all of the releases were evaluated as ground level releases. As such the release height used in PAVAN was 10.0 m, as specified in Table 3.1 of NUREG/CR-2858. The building wake term in PAVAN was set to 2011 m^2 based on the minimum cross-sectional area of the containment building. In addition, the licensee took credit for use of the calm wind speed array and the default terrain correction factors allowed by PAVAN.

The licensee used the following conservative assumptions in the determination of the offsite χ/Q values: (1) PAVAN was used to calculate the χ/Q values, (2) the higher of either the maximum sector χ/Q value or the 5-percent overall site x/Q value was selected for each time period, (3) the minimum distance to the site boundary was used in each downwind sector, (4) only ground level releases were used, (5) a conservative value for building wake was used, and (6) no downwind sectors that extended over Lake Michigan were excluded. The resulting χ/Q values are presented in Table 3 below, in addition to being summarized in Table 1.8.2-1, “Offsite Atmospheric Dispersion (χ/Q values) Factors for Analysis Events,” NAI Report No. NAI-1149-027 Revision 1, of the PNP LAR for AST implementation.

In conclusion, the NRC staff performed independent confirmatory analyses of the licensee’s assessments of the DBA offsite χ/Q values for the EAB and the LPZ. The NRC staff used the licensee’s meteorological data as input to PAVAN to perform the confirmatory analyses. On the basis of this review, the NRC staff concludes that the licensee’s EAB and LPZ χ/Q values for the DBA releases to the PNP’s EAB and LPZ as presented in Table 3 below and in Table 1.8.2-1, NAI Report No. NAI-1149-027, Revision 1, of the LAR for AST implementation, are acceptable for use in the DBA dose assessments performed in support of this proposed LAR for AST implementation.

3.3 Technical Specification Changes

3.3.1 TS Definitions Section 1.1, "Dose Equivalent I-131"

The intent of the TS on RCS specific activity is to ensure that assumptions made in the DBA radiological consequence analyses remain bounding. As such, the specification should have a basis consistent with the basis of the dose analyses. The licensee currently calculates DEI using thyroid DCFs, since the limiting analysis result was the thyroid dose. The AST analyses, however, determine the TEDE, rather than the whole-body dose and thyroid dose as done previously.

Typically, changes to the DEI definition are included as a part of the AST submittal. It is appropriate for those plants using the AST methodology to use a definition of DEI, based on the CEDE DCFs instead of thyroid DCFs. Some licensees using the AST methodology have chosen to reference the committed dose equivalent (CDE) thyroid dose in their TS definition of DEI. Although technically it is more accurate to reference the CEDE DCFs, the numerical difference in the calculated value of DEI using either CEDE or CDE thyroid DCFs, for a given isotopic mixture, is not significant.

The licensee has proposed to maintain the current definition of DEI, which is based on the determination of the inhalation dose to the thyroid. The licensee has proposed to use the DCFs from Federal Guidance Report No.11 (FGR No. 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submission, and Ingestion," Enviromental Protection Agency, 1998, to calculate DEI. The NRC staff has evaluated the proposed definition of DEI and has determined that the incorporation of either the CDE thyroid DCFs or the CEDE DCFs from FGR No.11 in the DEI definition is acceptable.

3.4 Maintaining Containment Sump pH

A variety of acids and bases are produced in the containment after a LOCA. The pH value of the containment sump will depend on the chemical species dissolved in the containment sump water. The licensee identified the following chemical species that are introduced into the containment sump in a post-LOCA environment: hydrochloric acid (HCl), nitric acid (HNO_3) and cesium hydroxide (CsOH). CsOH enters the containment directly from the RCS. HCl is produced by radiolytic decomposition of cable jacketing and HNO_3 is synthesized in the radiation field existing in the containment. The resultant containment sump pH will depend on the relative concentrations of these species and on the buffering action of sodium tetraborate (STB).

Maintaining sump water in a neutral or alkaline condition is needed to prevent dissolved radioactive iodine from being released to the containment atmosphere during the recirculation containment spray injection. Most of the iodine leaves the damaged core in an ionic form that is readily dissolved in the sump water. However, in an acidic environment, some of it becomes converted into elemental form that is much less soluble, causing re-evolution of iodine to the containment atmosphere. According to NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," the iodine entering the containment is at least 95-percent CsI with the remaining 5-percent as elemental and organic iodine plus hydriodic acid, with not less than 1-percent of each as iodine and hydriodic acid. In order to prevent release of elemental iodine to the containment atmosphere after a LOCA, the sump pH has to be maintained equal or higher than 7.

After a LOCA, the containment sump is mostly filled with water coming from the systems containing boric acid: SIRWT, Safety Injection Accumulators, and the RCS. This, in effect, will cause the sump water pH to become acidic. In order to keep the pH above 7, the licensee uses a buffer to maintain the pH above 7 for the 30-day period after LOCA.

The licensee performed a parametric analysis to determine the amount of STB required to maintain the pH above 7 for the 30-day period after a LOCA. The analysis was performed as a function of the quantity of borated water in the sump, the boron concentration in the sump, the sump water temperature and the desired equilibrium pH value. The analysis included computation of the formation of different boron species, water dissociation, hydrochloric and nitric acid formation from radiolysis, and iodine and cesium addition from core inventory.

The NRC staff reviewed the licensee's methodology, assumptions, and performed hand calculations to verify the resulting pH value after 30 days. The NRC staff's independent verification demonstrated the containment sump pH would remain above 7 for at least 30 days, consistent with the licensee's submittal.

Based on the review of the licensee's analyses and the NRC staff's independent verifications, the NRC staff concludes the licensee's analysis of the containment sump pH after a LOCA to be acceptable.

3.5 Commitments

Several of the AST analyses assumptions discussed herein are based on commitments. The implementation of these commitments is necessary to support the assumptions in this SE. The commitments are as follows:

3.5.1 Alternate buffer program

In the letter dated September 25, 2006, the licensee restated a commitment to implement a buffer program to maintain a pH of 7.0 - 8.0 post-LOCA with recirculation, during the 2007 fall refueling outage at PNP.

The commitment to implement a buffer program to maintain a pH of 7.0 - 8.0 post-LOCA with recirculation is necessary to prevent iodine re-evolution from the sump that has not been analyzed for in the AST LOCA analysis.

3.5.2 Plant modifications

3.5.2.1 Replacement of ECCS pump minimum flow recirculation isolation valves.

The licensee stated this commitment in the letter dated September 25, 2006. This modification, to reduce the back-leakage through recirculation valves CV-3027 and CV-3056, is required to support the recirculation line leakage assumption used in the calculation of the dose consequence resulting from SIRWT back-leakage as a component of the LOCA analyses.

3.5.2.2 Replacement of the CR normal air intake and purge dampers.

The licensee stated this commitment in the letter dated September 25, 2006. This modification, to replace CR HVAC dampers D-1, D-2, D-8, D-9, D-15, and D-16, is required to support the unfiltered inleakage assumptions, after isolation, as used in the CR habitability analyses.

The AST LAR dated September 25, 2006, included a commitment to install an alternate power source to allow the cross-tie of the LPSI suction piping. In a letter dated June 15, 2007, the licensee revised this commitment because it was determined that the cross-tie of the low pressure safety injection piping can be achieved procedurally, as opposed to requiring a physical plant modification. The commitment was revised to read:

3.5.2.3 ENO will modify plant emergency operating procedures to allow the cross-tie of the low pressure safety injection suction piping post LOCA following recirculation.

This modification of plant emergency operating procedures is necessary to support the assumption that, at 2 hours post-LOCA, an operator action is credited to cross-tie the LPSI suction headers to eliminate back-leakage through the SIRWT discharge lines.

3.5.3 Post-modification and periodic testing

In the letter dated September 25, 2006, Palisades has committed to conduct testing including: both post-modification and periodic tracer gas testing to verify CR unfiltered inleakage assumptions; verification of the timing of the operator action to cross-tie the LPSI suction headers; and any other necessary testing, following the implementation of the plant modifications described above, to validate that the modified plant configuration supports the assumptions used in the dose consequence analyses supporting the AST LAR.

3.5.4 Fuel Management

The licensee has committed in the September 21, 2007 letter, to updating the COLR by adding 5 rod power restrictions that were originally submitted in the September 7, 2007, letter.

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitments are best provided by the licensee's administrative processes, including its commitment management program (see Regulatory Issue Summary 2000-017, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff"). The above regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

4.0 SUMMARY

Several of the AST analyses assumptions discussed herein are based on commitments. The implementation of these commitments is necessary to support the assumptions in this SE. As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of DBAs with full implementation of an AST at PNP. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The NRC staff also finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses will comply with these criteria. The NRC staff further finds reasonable assurance that PNP as modified by this license amendment, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of DBAs.

After an accident, the pH of the containment sump water is determined by the amounts of acidic and basic chemical materials either released from the damaged core or generated in containment and subsequently dissolved in the sump water. It is important to control the sump pH because if it falls below 7, radioactive iodine could be released to the containment atmosphere. The addition of a buffering agent such as STB will keep the water pH above 7, therefore preventing the iodine from being released. The licensee's analysis has indicated that containment sump water will remain greater than 7 for at least 30 days. The NRC staff reviewed the licensee's methodology for determining pH and performed an independent evaluation of the licensee's calculations. Based on its evaluation, the NRC staff concludes that the licensee's proposed actions will maintain the sump water pH greater than 7 for 30 days following a LOCA, thus preventing the release of radioactive iodine into the containment atmosphere.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the PNP design basis is superseded by the AST proposed by the licensee. The previous offsite and CR accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR Part 50.67, or fractions thereof, as defined in RG 1.183. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined the PNP design basis, and modified by the present amendment.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment. The Michigan State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (72 FR 8804). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, 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.

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Table 1
PNP Radiological Consequences Expressed as TEDE [1]
(rem)

Design Basis Accidents	EAB [2]	LPZ [3]	CR
Loss of Coolant Accident	1.3E+01	3.3E+00	4.0E+00 [9]
Main steamline break accident [4]	6.7E−01	2.0E−01	3.3E+00 [11]
Steam Generator Tube Rupture [5]	9.9E−01	2.2E−01	3.8E+00 [10]
Dose acceptance criteria	2.5E+01	2.5E+01	5.0E+00
Steam Generator Tube Rupture [6]	1.2E+00	2.1E−01	3.5E+00 [10]
Small Line Break Outside Containment	4.1E−01	5E−02	5.3E−01 [10]
Dose acceptance criteria	2.5E+00	2.5E+00	5.0E+00
Control Rod Ejection Accident [7]	2.7E+00	4.3E−01	1.1E+00 [9]
Control Rod Ejection Accident [8]	2.6E+00	6.8E−01	1.1E+00 [9]
Dose acceptance criteria	6.3E+00	6.3E+00	5.0E+00
Fuel Handling Accident (FHA) in containment	2.2E+00	2.8E−01	4.0E+00 [10]
FHA in Fuel Building - 10-percent filtration	2.0E+00	2.5E−01	3.7E+00 [10]
FHA in Fuel Building - 34-percent filtration	1.6E+00	2.0E−01	2.8E+00 [10]
FHA in Fuel Building - 50-percent filtration	1.3E+00	1.7E−01	2.2E+00 [10]
Dose acceptance criteria	6.3E+00	6.3E+00	5.0E+00
Spent Fuel Cask Drop Accident			
30 day decay - 90-percent fuel building filtration	2.0E+00	2.5E−01	1.4E+00 [10]
30 day decay - 82.5-percent fuel building filtration	2.8E+00	3.5E−01	2.0E+00 [10]
90 day decay - no CR isolation	8E−02	1E−02	1.7E+00 [10]
Dose acceptance criteria	6.3E+00	6.3E+00	5.0E+00

[1] Total effective dose equivalent
[2] Exclusion area boundary
[3] Low population zone
[4] Assumes 2-percent of the fuel experiences DNB
[5] Pre-accident iodine spike
[6] Concurrent iodine spike
[7] Assumes containment release
[8] Assumes secondary side release
[9] Assumes 10 cfm CR unfiltered inleakage
[10] Assumes 100 cfm CR unfiltered inleakage
[11] Assumes 20 cfm CR unfiltered inleakage

Note: Licensee results are expressed to a limit of two significant figures

Table 2 (Page 1 of 3)
PNP Control Room Atmospheric Dispersion Factors

Source Location / Duration	χ/Q (sec/m^3)
For evaluating unfiltered intake from closest containment point to CR normal intake B	
0 - 2 hours	1.43E−02
2 - 8 hours	1.11E−02
8 - 24 hours	4.13E−03
24 - 96 hours	3.23E−04
96 - 720 hours	2.49E−03
For evaluating filtered intake from closest containment point to CR emergency intake	
0 - 2 hours	7.26E−04
2 - 8 hours	6.18E−04
8 - 24 hours	2.47E−04
24 - 96 hours	1.77E−04
96 - 720 hours	1.30E−04
For evaluating unfiltered intake from PNP stack to CR normal intake B	
0 - 2 hours	6.10E−03
2 - 8 hours	4.32E−03
8 - 24 hours	1.73E−03
24 - 96 hours	1.27E−03
96 - 720 hours	9.79E−04
For evaluating filtered intake from PNP stack to CR emergency intake	
0 - 2 hours	8.32E−04
2 - 8 hours	7.69E−04
8 - 24 hours	2.83E−04
24 - 96 hours	2.15E−04
96 - 720 hours	1.57E−04
Safety injection refueling water tank (SIRWT) to CR normal intake B	
0 - 2 hours	9.57E−02
2 - 8 hours	7.59E−02
8 - 24 hours	2.87E−02
24 - 96 hours	2.19E−02
96 - 720 hours	1.65E−02
SIRWT to CR emergency intake	
0 - 2 hours	9.66E−04
2 - 8 hours	7.92E−04
8 - 24 hours	3.13E−04
24 - 96 hours	2.20E−04
96 - 720 hours	1.64E−04

Table 2 (Page 2 of 3)
PNP Control Room Atmospheric Dispersion Factors

Source Location / Duration		χ/Q (sec/m^3)
For evaluating unfiltered intake from PNP equipment hatch to CR normal intake B		
0 - 2 hours		1.25E−02
2 - 8 hours		9.83E−03
8 - 24 hours		3.62E−03
24 - 96 hours		2.86E−03
96 - 720 hours		2.28E−03
For evaluating filtered intake from PNP equipment hatch to CR emergency intake		
0 - 2 hours		7.32E−04
2 - 8 hours		6.13E−04
8 - 24 hours		2.45E−04
24 - 96 hours		1.75E−04
96 - 720 hours		1.29E−04
For evaluating unfiltered intake from TB NE roof exhaust to CR normal intake A		
0 - 2 hours		1.31E−02
2 - 8 hours		1.13E−02
8 - 24 hours		4.68E−03
24 - 96 hours		2.87E−03
96 - 720 hours		2.36E−03
For evaluating filtered intake from NW & NE TB roof exhaust to CR emergency intake		
0 - 2 hours	NW exhaust	7.99E−04
2 - 8 hours	NW exhaust	6.43E−04
8 - 24 hours	NE exhaust	2.58E−04
24 - 96 hours	NW exhaust	1.75E−04
96 - 720 hours	NW exhaust	1.32E−04
For evaluating unfiltered intake from closest ADV to CR normal intake A		
0 - 2 hours		1.65E−02
2 - 8 hours		1.34E−02
8 - 24 hours		5.40E−03
24 - 96 hours		4.03E−03
96 - 720 hours		2.98E−03
For evaluating unfiltered intake from closest ADV to CR emergency intake		
0 - 2 hours		7.36E−04
2 - 8 hours		6.42E−04
8 - 24 hours		2.43E−04
24 - 96 hours		1.75E−04
96 - 720 hours		1.28E−04

Table 2 (Page 3 of 3)
PNP Control Room Atmospheric Dispersion Factors

Source Location / Duration		χ/Q (sec/m³)
For evaluating unfiltered intake from closest SSRV to CR normal intake A & B		
0 - 2 hours	intake B	2.11E−02
2 - 8 hours	intake B	1.71E−02
8 - 24 hours	intake B	6.91E−03
24 - 96 hours	intake A	4.98E−03
96 - 720 hours	intake A	3.72E−03
For evaluating filtered intake from closest SSRV to CR emergency intake		
0 - 2 hours		7.96E−04
2 - 8 hours		6.91E−04
8 - 24 hours		2.60E−04
24 - 96 hours		1.90E−04
96 - 720 hours		1.37E−04
For evaluating MSLB to CR normal intake A		
0 - 2 hours		2.20E−02
2 - 8 hours		1.75E−02
8 - 24 hours		7.10E−03
24 - 96 hours		5.24E−03
96 - 720 hours		3.87E−03
For evaluating MSLB to CR emergency intake		
0 - 2 hours		8.65E−04
2 - 8 hours		7.56E−04
8 - 24 hours		2.81E−04
24 - 96 hours		2.04E−04
96 - 720 hours		1.47E−04

Table 3
Offsite Atmospheric Dispersion Factors (sec/m^3)

Receptor/ Source Location / Duration		χ/Q (sec/m^3)
EAB		
	0 - 2 hours	5.39E−04[1]
	0 - 8 hours	3.31E−04
	8 - 24 hours	2.59E−04
	24 - 96 hours	1.53E−04
	96 - 720 hours	7.14E−05
LPZ		
	0 - 2 hours	6.66E−05
	0 - 8 hours	3.03E−05
	8 - 24 hours	2.04E−05
	24 - 96 hours	8.67E−06
	96 - 720 hours	2.54E−06

[1] Note that the 0 - 2 hour EAB χ/Q factor was used for the entire evaluation period.

Table 4
PNP Control Room Data, Assumptions and LOCA direct shine results

Control room envelope net volume		76,451 ft3
CR volume		35,923 ft3
CR normal mode unfiltered makeup flow rate or infiltration		
	LOCA	384.2 cfm infiltration
	FHA, MSLB, SGTR, SLBOC	660 cfm makeup flow
	CRE (0 - 90 seconds)	384.2 cfm infiltration
	CRE (90 seconds - 20 minutes)	660 cfm makeup flow
	Spent fuel cask drop case 1 & 2	N/A - emergency mode only
	Spent fuel cask drop case 3	660 cfm makeup flow
CR emergency mode filtered makeup flow rate		1413.6 cfm
CR emergency mode filtered recirculation flow rate		1413.6 cfm

CR emergency mode initiation time and unfiltered inleakage		Initiation	Inleakage
	LOCA	90 seconds	10 cfm
	FHA	20 minutes	100 cfm
	MSLB	20 minutes	10 cfm
	SGTR	20 minutes	100 cfm
	SLBOC	20 minutes	100 cfm
	CRE	20 minutes	
	Spent fuel cask drop case 1 & 2	T = 0 seconds	100 cfm
	Spent fuel cask drop case 3	N/A	660 cfm

CR emergency mode credited filter efficiencies		
	Elemental iodine	99-percent
	Organic iodine	99-percent
	Particulates	99-percent
CR operator breathing rate		
	0 - 720 hours	3.5E-04 m3/sec
CR occupancy factors		
	0 - 24 hours	1.0
	24 - 96 hours	0.6
	96 - 720 hours	0.4
LOCA CR direct shine dose		
	Source	Direct shine dose (rem)
	Containment	0.028
	CR filters	0.005
	Purge line	0.000
	External cloud	0.232
	SIRWT	1.269

Table 5 (Page 1 of 2)
PNP Data and Assumptions for the LOCA

Power level for core inventory calculations	2703 Mwt
Power level for thermal hydraulic calculations	2580.6 MWt
Limiting power level for AST LOCA evaluation	2580.6 MWt
Concurrent LOOP	Yes
Primary containment free volume	1.64E+06 ft^3
Primary containment leak rate	
0 - 24 hours	0.1 weight percent/day
24 - 720 hours	0.05 weight percent/day
Elemental iodine spray removal coefficients	
0.0 - 0.016667 hours	0.0 hr^{-1}
0.016667 - 2.515 hours	4.8 hr^{-1}
2.515 - 720 hours	0.0 hr^{-1}
Particulate spray removal coefficients	
0.0 - 0.01667 hours	0.0 hr^{-1}
0.016667 - 3.385 hours	1.8 hr^{-1}
3.385 - 10.0 hours	0.18 hr^{-1}
10.0 - 720 hours	0.0 hr^{-1}
Elemental iodine wall deposition coefficient (0 -720 hours)	2.3 hr^{-1}
Particulate natural deposition removal coefficient	
0.0 - 10.0 hours	0.0 hr^{-1}
10.0 - 720 hours	0.1 hr^{-1}
IIodine chemical form in containment atmosphere	
cesium iodide	95-percent
elemental iodine	4.85-percent
organic iodine	0.15-percent
Minimum sump volume	39,054 ft^3 (292,143 gallons)
Containment sump pH	$\geq$ 7
Assumed ECCS leakage (two times TS limit of 0.2 gpm)	0.053472 ft^3/min (0.4 gpm)
Assumed ECCS leakage start time	19 minutes
Flashing fraction	
Calculated	0.03
Assumed	0.1
Chemical form of released iodine from ECCS leakage	
elemental	97-percent
organic	3-percent

Table 5 (Page 2 of 2)
PNP Data and Assumptions for the LOCA

IIodine DF for ECCS leakage (current design basis)	2
IInitial SIRWT liquid inventory	4,144 gallons
SIRWT back-leakage rates	
0 - 0.3167 hrs (19 minutes)	0.0 gpm
0.31667 - 2 hrs	4.4 gpm
2 - 720 hrs	0.05 gpm
Time dependent SIRWT pH values	
Selected times in hours	SIRWT pH
0.00	4.500
8.00	4.546
24.00	4.550
96.00	4.570
720.00	4.711
Time dependent SIRWT total iodine concentration (gm-atom/liter)	
Selected times in hours	SIRWT Iodine concentration
0.00	0.00E+00
8.00	5.02E-06
24.00	5.48E-06
96.00	7.46E-06
720.00	1.93E-05
Time dependent SIRWT liquid temperature	
Selected times in hours	Temperature (°F)
0.00	100.0
8.00	101.3
24.00	103.2
96.00	105.0
720.00	104.4
Time dependent SIRWT elemental iodine fraction	
Selected times in hours	Elemental iodine fraction
0.00	0.00E+00
8.00	8.16E-02
24.00	8.68E-02
96.00	1.06E-01
720.00	1.49E-01

Table 6
PNP Data and Assumptions for the FHA

Power level for core inventory calculations	2703 MWt
Minimum post shutdown fuel handling time (decay time)	48 hours
Number of fuel assemblies in core	204
Number of failed rods for fuel handling accident	All pins in 1 fuel assembly
Core radial peaking factor	2.04
Minimum pool water depth	22.5 feet
Fuel clad damage gap fractions	
I-131	8-percent
Remainder of halogens	5-percent
Kr-85	10-percent
Remainder of noble gases	5-percent
Alkali metals	12-percent
Pool DF	
Noble gases and organic iodine	1
Aerosols	Infinite
Elemental iodine (22.5 ft of water cover)	252
Overall iodine (22.5 ft of water cover)	183.07 (effective DF)
Chemical form of iodine in pool	
Elemental	99.85-percent
Organic	0.15-percent
Duration of release to the environment	2 hours
FHB Ventilation filter efficiencies	
Elemental iodine	94-percent
Organic iodine	94-percent
Particulate and aerosol activity	Retained in cavity and SFP water
CR manual isolation and emergency mode initiation time	20 minutes
CR unfiltered air intake prior to isolation	660 cfm
CR unfiltered inleakage in emergency mode	100 cfm

Table 7
PNP Data and Assumptions for the MSLB Accident

Power level for core inventory calculations	2703 MWt
Power level for thermal-hydraulic calculations	2580.6 MWt
Core average burnup	39,300 MWD/MTU
Primary coolant system (PCS) mass	432,977 lbm
Steam generator secondary side mass	
Maximum (hot standby) for faulted SG release	210,759 lbm
Minimum (full power) for intact SG release	141,065 lbm
Radial peaking factor	2.04
Fuel damage	
DNB	0.5-percent
Fuel centerline melt	0
Steam generator tube leakage rate	0.3 gpm per SG
Time to establish shutdown cooling terminating steam release	8 hours
Time for PCS to reduce to 212 °F terminating SG tube leakage	12 hours
Release from faulted SG	Instantaneous
Steaming from intact SGs	
0-8 hours	800,000 lbm
8-720 hours	0
Secondary coolant iodine activity	0.1 µCi/gm DE I-131
SG secondary side partition coefficients	
Faulted SG	1 (none)
Intact SGs	100
CR manual isolation and emergency mode initiation time	20 minutes
CRE unfiltered air intake prior to isolation	660 cfm
CRE unfiltered inleakage in emergency mode	20 cfm

Table 8
PNP Data and Assumptions for the SGTR Accident

Power level for core inventory calculations	2307 MWt
Power level for thermal-hydraulic calculations	2580.6 MWt
Initial PCS equilibrium activity	
DE I-131	1.0 µCi/gm
Gross beta & gamma activity	100/E-bar
Initial secondary side equilibrium DE I-131 activity	0.1 µCi/gm
Maximum pre-accident spike iodine concentration	40 µCi/gm DE I-131
Duration of accident initiated iodine spike	8 hours
Steam generator tube leakage rate	0.3 gpm per SG
Time to establish shutdown cooling terminating steam release	8 hours
Primary coolant system mass	
Pre-accident iodine spike case	529,706 lbm
Concurrent iodine spike case	459,445 lbm
Secondary side minimum mass per steam generator	141,065 lbm

SG secondary side PCs, FFs and DFs		
Intact SGs	PC = 100	
Ruptured SG (non-flashed tube flow)	PC = 100	
Ruptured SG (flashed tube flow)	FF	DF
0 - 707.1 seconds	0.110	1.0
707.1 - 736 seconds	0.065	1.0
736 - 859 seconds	0.031	1.002299
859 - 1,090 seconds	0.023	1.045037
1,090 - 1,800	0.006	1.452436
1,800 - 13,100	0.006	1.467378
13,100 - 28,800	0.006	58.16008

SGTR integrated mass release (lbm)

Time hours	Break flow	Steam release (ruptured SG)	Steam release (intact SG)
0 - 0.196417	24,011.15	0	0
0.196417 - 0.5	37,111.85	44,654	53,574
0.5 - 1.388889	81,281	22,152.3	109,629.6
1.388889 - 2	40,798	15,229.7	75,370.4
2 - 3.638889	64,773	75,485.6	145,983.5
3.638889 - 8	357,126	200,868.4	388,464.5
8 - 720	0	0	0

Concurrent (335x) iodine spike appearance rate

Isotope	Appearance rate (Ci/min)	Time of depletion (hours)
Iodine 131	58.0966961	> 8
Iodine 132	79.8319317	> 8
Iodine 133	90.1310904	> 8
Iodine 134	74.0318685	> 8
Iodine 135	68.9790622	> 8

Table 9
PNP Data and Assumptions for the SLBOC Accident

Power level for core inventory calculations		2703 MWt
PCS equilibrium activity		
	DE I-131	1.0 µCi/gm
	Gross beta & gamma activity	100/E-bar
Break flow rate		160 gpm
Break temperature		135 °F
Break pressure		35 psia
Time required to isolate break		60 minutes
Iodine fraction released from break flow		10-percent
Reactor building ventilation system ventilation		not credited

Concurrent (500x) iodine spike appearance rate

Isotope	Appearance rate (Ci/min)
Iodine 131	86.7114868
Iodine 132	119.152137
Iodine 133	134.524016
Iodine 134	110.495326
Iodine 135	102.953824

Table 10
PNP Data and Assumptions for the CRE Accident

Power level for core inventory calculation	2703 MWt
Power level for thermal-hydraulic calculations	2580.6 MWt
Core average fuel burnup	39,300 MWD/MTU
Fuel enrichment	3.0 - 5.0
Maximum radial peaking factor	2.04
Containment volume	1.64E+06 ft^3
Containment leak rate	
0 to 24 hours	0.10% (by weight)/day
24 - 720 hours	0.05% (by weight)/day
Containment natural deposition coefficients	
Aerosols	0.1 hr^{-1}
Elemental iodine	1.3 hr^{-1}
Organic iodine	not credited
Fuel DNB	14.7%
Fuel centerline melt	0.5%
Initial PCS equilibrium activity	
DE I-131	1.0 µCi/gm
Gross beta & gamma activity	100/E-bar
Initial secondary side equilibrium iodine activity	0.1 µCi/gm
SG secondary side partition coefficient	100
SG tube leakage	0.3 gpm per SG
Time to establish shutdown cooling	8 hours
PCS mass	432,976.8 lbm
SG secondary side mass	141,065 lbm per SG

Chemical form of released iodine	To containment	From SGs
Particulate	95%	0%
Elemental	4.85%	97%
Organic	0.15%	3%

Control rod ejection steam release (lbm)	
0 - 1,100 seconds	107,158.8
1,100 seconds - 0.5 hours	31,336.8
0.5 - 8 hours	1,007,100
> 8 hours	0

Table 11
PNP Data and Assumptions for the Spent Fuel Cask Drop

Core thermal power level before shutdown	2703 MWt
Core average burnup	39,300 MWD/MTU
Discharged fuel assembly burnup	39,000 - 58,900 MWD/MTU
Delay prior to cask drop (decay time)	
Cases 1 & 2	30 days
Case 3	90 days
Number of fuel assemblies in core	204
Number of fuel assemblies damaged in cask drop	73
Core radial peaking factor	2.04
Water level above damaged fuel	23.4 feet
Fuel clad damage gap fractions	
I-131	8%
Remainder of halogens	5%
Kr-85	10%
Remainder of noble gases	5%
Alkali metals	12%
Pool DF	
Noble gases and organic iodine	1
Aerosols	Infinite
Elemental iodine (23.4 ft of water cover)	285
Overall iodine (23.4 ft of water cover)	200 (effective DF)
Chemical form of iodine in pool	
Elemental	99.85%
Organic	0.15%
Duration of release	2 hours

Palisades Plant

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