NEXTera ENERGY
POINT BEACH

May 8, 2009

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10 CFR 50.90

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

Supplement to License Amendment Request 241
Regulatory Guide 1.183 Compliance Matrix

Reference: (1) FPL Energy Point Beach, LLC, License Amendment Request 241, dated December 8, 2008, Alternative Source Term (ML083450683)

NextEra Energy Point Beach, LLC (NextEra) has enclosed a Regulatory Guide 1.183 Compliance Matrix. This matrix was discussed during a teleconference held with NRC staff on January 13, 2009. The enclosure supports the review of the NextEra submittal of License Amendment Request 241, "Alternative Source Term."

This supplement does not affect the no significant hazards consideration provided in Reference 1.

This letter contains no new commitments and no revisions to existing commitments.

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

I declare under penalty of perjury that the foregoing is true and correct.
Executed on May 8, 2009.

Very truly yours,

NextEra Energy Point Beach, LLC

Larry Meyer
Site Vice President

Enclosure

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

ENCLOSURE

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

SUPPLEMENT TO LICENSE AMENDMENT REQUEST 241
REGULATORY GUIDE 1.183 COMPLIANCE MATRIX

55 pages follow

Regulatory Guidance	Comments
3. ACCIDENT SOURCE TERM	
Fission Product Inventory The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty[8]. 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[9]. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID 14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels. For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.	Complies The source terms derived from the reactor core fission product inventory are listed in Table 5 of LAR 241, Enclosure 3. The inventory of the nuclides in the reactor core is based on maximum full-power operation of the core at a power level equal to 1811 MWt, which includes a 0.6% power uncertainty. Further, the calculated nuclide inventories are increased by a factor of 1.04 to account for potential fuel variations (LAR 241, Enclosure 3, Section 3.1). The period of irradiation was of sufficient duration to achieve equilibrium or maximum isotopic concentrations, for a representative operating cycle. The core inventory was determined with the ORIGEN-S code (LAR 241, Enclosure 3, Sections 1.5 and 3.1). For DBA events that do not involve the entire core (Locked Rotor Accident (LR), Control Rod Ejection Accident (CRDE), and Fuel Handling Accident (FHA)), a peaking factor of 1.7 is applied to the average assembly inventory (LAR 241, Enclosure 3, Tables 20, 22 and 23). For the Design Basis (DBA) Loss of Coolant Accident (LOCA), all fuel assemblies in the core are assumed to be affected, and the total core inventory is used (LAR 241, Enclosure 3, Table 5). No adjustments for less than full power were made. For the FHA, a decay time of 65 hours from the time of shutdown was assumed (LAR 241, Enclosure 3, Table 23).

No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled. Footnote 8 - The uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02. Footnote 9 - Note that for some radionuclides, such as Cs-137, equilibrium will not be reached prior to fuel offload. Thus, the maximum inventory at the end of life should be used.	

3.2 Release Fractions [10, 11] The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. Footnote 10 - The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel. Footnote 11- The release fractions listed here 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 GWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case- by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.	Complies For the LOCA, release fractions are as given in Table 2 of RG 1.183 and are applied to the core inventory described in Section 3.1 above. For non-LOCA events, see gap inventory discussion provided in LAR 241, Enclosure 3, Section 3.3. The PBNP core consists of fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material. PBNP does not use MOX fuel. Does Not Comply with Footnote 11 Some fuel was determined to exceed the burnup criteria of Footnote 11. Alternate gap fractions were assumed for this fuel (LAR 241, Enclosure 3, Section 3.3). Also, see FHA in Appendix B of this document.

3.3 Timing of Release Phases Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase[12]. For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage. For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used. Footnote 12 - In lieu of treating the release in a linear ramp manner, the activity for each phase can be modeled as being released instantaneously at the start of that release phase, i.e., in step increases.	Complies The activity released from the core during each release phase is modeled in a linear fashion with the RADTRAD code (LAR 241, Enclosure 3, Section 1.5). Gap release and early-in-vessel release are modeled in the RADTRAD release fraction and timing file (rft). The core fraction specified in the rft file is released linearly during the specified time. The non-LOCA activity release rate is instantaneous, consistent with RG 1.183. The non-LOCA events that assume fuel damage include the CRDE, LR and FHA. Instantaneous release (10^{-6} hours) is shown in the respective rft files for the CRDE and LR. The FHA release to the containment occurs over 2 hours (LAR 241, Enclosure 3, Table 23). The RVHD activity release rate is also instantaneous (10^{-6} hours). The RVHD is specific to PBNP; not a RG 1.183 event. The leak-before-break methodology is not credited in the PBNP AST analyses. As such, the onset of the gap release is delayed by 30 seconds, consistent with RG 1.183, Table 4 (LAR 241, Enclosure 3, Table 16).
3.4 Radionuclide Composition Table 5 lists the elements in each radionuclide group that should be considered in DBAs.	Complies The elements assumed in each radionuclide group are consistent (not identical) with those of Table 5 of RG 1.183 (LAR 241, Enclosure 3, Table 5). The specific nuclide selection was performed by Westinghouse. The nuclide grouping is consistent with that used in the RADTRAD users guide (NUREG/CR-6604).

3.5 Chemical Form Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95% of the iodine released should be assumed to be cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	Complies The assumed iodine chemical forms are consistent with RG 1.183 (LAR 241, Enclosure 3, Tables 18 through 23). Within this document, refer to the following Appendices: Appendix A, for LOCA Appendix B for FHA Appendix E for MSLB Appendix F for SGTR Appendix G for LR Appendix H for CRDE For the RVHD, the assumed chemical form of iodine released is 100% elemental. The RVHD is not addressed in RG 1.183 (LAR 241, Enclosure 3, Table 24).
3.6 Fuel Damage in Non-LOCA DBAs The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.	Complies For the FHA (1 fuel assembly) and CRDE (10% of core), fuel damage assumptions are based on the current licensing basis cases that result in conservative radioactivity releases (LAR 241, Enclosure 3, Tables 22 and 23). The CLB values are provided in PBNP FSAR Chapter 14. Applicable FSAR sections are listed in LAR 241, Enclosure 3. The locked rotor analysis assumes 30% failed fuel (LAR 241, Enclosure 3, Table 20). The locked rotor rods in DNB methodology is described in LAR 241, Enclosure 3, Section 6.3.

4. DOSE CALCULATION METHODOLOGY	
4.1 Offsite Dose Consequences The following assumptions should be used in determining the TEDE for persons located at or beyond the boundary of the exclusion area (EAB):	
4.1.1 The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences and the released radioactivity[13]. Footnote 13 - The prior practice of basing inhalation exposure on only radioiodine and not including radioiodine in external exposure calculations is not consistent with the definition of TEDE and the characteristics of the revised source term.	Complies The dose calculations determine the TEDE dose, with all significant progeny included, as the sum of the CEDE and the EDE (LAR 241, Enclosure 3, Section 2.2). TEDE doses were calculated with RADTRAD Version 3.03 which implements the TEDE dose methodology (LAR 241, Enclosure 3, Section 1.5).

4.1.2 The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE. The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	Complies The dose conversion factors (DCF's) used in determining the CEDE dose are from EPA Federal Guidance Report 11. The CEDE dose factors are provided in LAR 241, Enclosure 3, Table 1.
4.1.3 For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	Complies The assumed offsite breathing rates are those specified in Section 4.1.3 of RG 1.183 (LAR 241, Enclosure 3, Table 3).
4.1.4 The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE.	Complies EDE is used to determine the submergence dose in a semi-infinite cloud. The assumed conversion factors are those of Federal Guidance Report 12 (LAR 241, Enclosure 3, Table 2).

4.1.5 The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67[14]. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6). Footnote 14 - With regard to the EAB TEDE, the maximum two-hour value is the basis for screening and evaluation under 10 CFR 50.59. Changes to doses outside of the two-hour window are only considered in the context of their impact on the maximum two-hour EAB TEDE.	Complies The TEDE is determined for the most limiting person for a two-hour period at the EAB and the maximum two-hour dose is reported. LAR 241, Enclosure 3, Section 2.3 states that doses at the EAB are for the worst 2-hour period. The EAB X/Q is determined for the worst sector. The combination of the maximum 2-hr dose using the worst case X/Q results in the dose to the limiting person.
4.1.6 TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Complies The LPZ TEDE is calculated up to the time releases are terminated (LAR 241, Enclosure 3, Section 2.3). The X/Q values for the LPZ are based on the worst sector.
4.1.7 No correction should be made for depletion of the effluent plume by deposition on the ground.	Complies No plume depletion due to ground deposition is credited (LAR 241, Enclosure 3, Section 2.2).

4.2 Control Room Dose Consequences The following guidance should be used in determining the TEDE for persons located in the control room:	
4.2.1 The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	Complies. The LOCA shine dose to CR personnel (due to containment and ECCS leakage) includes doses from containment and the auxiliary building, and doses from contained sources and the external plume (LAR 241, Enclosure 3, Section 6.1). The LOCA 30-day DDE is 0.28 rem (LAR 241, Enclosure 3, Section 6.1). The non-LOCA CR TEDE doses all have sufficient margin to the 5 rem limit (LAR 241, Enclosure 3, Section 1.2).

4.2.2 The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.	Does not Comply The control room TEDE doses are determined using the same source term, in-plant transport, and release assumptions used for determining the EAB and the LPZ TEDE values, resulting in conservative results for the control room. However, the offsite and CR atmospheric transport assumptions differ. CR X/Q values assume meteorological data from years 2000 through 2005, while the offsite CLB X/Q values use data from years 1991 through 1993 (LAR 241, Enclosure 3, Section 4.4). PBNP has assessed the CLB X/Q values against X/Q values generated using the PAVAN computer code with the meteorological data collected at PBNP from September 2000 to September 2005 and determined the CLB X/Q's are conservative (LAR 241, Enclosure 3, Section 4.4).
4.2.3 The models used to transport radioactive material into and through the control room[15], and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel. Footnote 15 - The iodine protection factor (IPF) methodology of Reference 22 may not be adequately conservative for all DBAs and control room arrangements since it models a steady-state control room condition. Since many analysis parameters change over the duration of the event, the IPF methodology should only be used with caution. The NRC computer codes HABIT (Ref. 23) and RADTRAD (Ref. 24) incorporate suitable methodologies.	Complies The models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, are conservative and consistent with the recommendations of RG 1.183. Doses due to airborne radioactivity were calculated with RADTRAD. The S&W PERC2 code was used to generate the source terms in the containment atmosphere, in the external plume passing the CR, and in the CR charcoal and HEPA filters to determine the shine dose for the CR operator (LAR 241, Enclosure 3, Section 1.5).

4.2.4 Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance. The control room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.	Complies Credit is taken for CR emergency intake and recirculation filtration (LAR 241, Enclosure 3, Table 4). The credited filters are qualified and acceptable per the PBNP Ventilation Filter Testing Program (PBNP TS 5.5.10). In addition, the FHA (LAR 241, Enclosure 3, Section 6.6) credits a high CR radiation alarm (RE-101 or RE-235) (LAR 241, Enclosure 3, Section 5.2) for initiating CREFS within 10 minutes of an FHA (LAR 241, Enclosure 3, Section 6.6).
4.2.5 Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Complies No credit is taken for the use of personal protective equipment or prophylactic drugs (LAR 241, Enclosure 3, Section 5.1).

4.2.6 The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days[16]. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second. Footnote 16 - This occupancy is modeled in the X/Q values determined in Reference 22 and should not be credited twice. The ARCON96 Code (Ref. 26) does not incorporate these occupancy assumptions, making it necessary to apply this correction in the dose calculations.	Complies The assumed breathing rates and occupancy factors for control room operator dose are those specified in Section 4.2.6 of RG 1.183 (LAR 241, Enclosure 3, Table 4).
4.2.7 Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. Equation 1 may be used to correct the semi-infinite cloud dose, DDEinfinity, to a finite cloud dose, DDEfinite, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22). DDEfinite = DDEinfinity * V0.338 / 1173 Equation 1	Complies Control room doses are calculated using dose conversion factors identified in Regulatory Position 4.1 above. Offsite and CR doses were calculated using RADTRAD (LAR 241, Enclosure 3, Section 1.5). CR and offsite TEDE doses are calculated with the DCFs provided in LAR 241, Enclosure 3, Tables 1 and 2.

4.3 Other Dose Consequences The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). 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. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Complies RG 1.183, Item 6, states that NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID14844 assumptions for performing the required EQ analyses. However, *no plant modifications are required to address the impact of the* difference in source term characteristics (i.e., AST vs. TID14844) on EQ doses pending the outcome of the evaluation of the generic issue. See Appendix I of this document.
4.4 Acceptance Criteria The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6 of RG 1.183. The acceptance criteria for the various NUREG-0737 (Ref. 2) items generally reference General Design Criteria 19 (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii). Footnote 17 - For PWRs with steam generator alternative repair criteria, different dose criteria may apply to steam generator tube rupture and main steam line break analyses.	Complies The appropriate regulatory limits of 10 CFR 50.67 and RG 1.183 are used. DBA analyses demonstrate that regulatory limits are not exceeded. Results are presented in terms of TEDE (LAR 241, Enclosure 3, Section 1.2). PBNP TS 5.5.8, "Steam Generator (SG) Program," describes alternative repair criteria. Table 6, "Accident Dose Criteria," was applied (LAR 241, Enclosure 3, Section 1.2).

5. ANALYSIS ASSUMPTIONS AND METHODOLOGY	
5.1 General Considerations	
5.1.1 Analysis Quality The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Many physical processes and phenomena are represented by conservative, bounding assumptions rather than being modeled directly. The staff has selected assumptions and models that provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion. Licensees should exercise caution in proposing deviations based upon data from a specific accident sequence since the DBAs were never intended to represent any specific accident sequence -- the proposed deviation may not be conservative for other accident sequences.	Complies The analyses have been prepared, reviewed and will be maintained in accordance with quality assurance programs that comply with 10 CFR Part 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The dose analyses are consistent with the recommendations of RG 1.183, with the exception of the deviations noted in LAR 241, Enclosure 3, Section 1.4.

5.1.2 Credit for Engineered Safeguard Features Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Complies Non-safety-related mitigation features are credited in the dose analyses. These systems include CREFS and VNPAB. These systems are designed to function following a single active failure (LAR 241, Enclosure 3, Section 1.4). In general, the LOOP was used to limit equipment availability. Where credited, safety related mitigating structures, systems, and components were assumed to operate consistent with single failure, design basis, emergency power, and other requirements.

5.1.3 Assignment of Numeric Input Values The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis. For example, assuming minimum containment system spray flow is usually conservative for estimating iodine scrubbing, but in many cases may be nonconservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by technical specifications, the value used in the analysis should be that specified in the technical specifications[18]. If a range of values or a tolerance band is specified, the value that would result in a conservative postulated dose should be used. If the parameter is based on the results of less frequent surveillance testing, e.g., steam generator nondestructive testing (NDT), consideration should be given to the degradation that may occur between periodic tests in establishing the analysis value. Footnote 18-Note that for some parameters, the technical specification value may be adjusted for analysis purposes by factors provided in other regulatory guidance. For example, ESF filter efficiencies are based on the guidance in Regulatory Guide 1.52 (Ref. 25) and in Generic Letter 99-02 (Ref. 27) rather than the surveillance test criteria in the technical specifications. Generally, these adjustments address potential changes in the parameter between scheduled surveillance tests.	Complies Conservative parameters are assumed when calculating each contributor in the dose analyses, consistent with the guidance of RG 1.183. Assumed analysis efficiencies for CREFS HEPA filters and adsorbers are consistent with the recommendations of RG 1.52, Rev. 3, Table 1 (LAR 241, Enclosure 3, Table 4).

5.1.4 Applicability of Prior Licensing Basis The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, "Backfitting." However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	Complies Proposed changes to the PBNP CLB, required to implement the AST, are described in LAR 241, Enclosure 3, Section 1.3.

5.2 Accident-Specific Assumptions The appendices to this regulatory guide provide accident-specific assumptions that are acceptable to the staff for performing analyses that are required by 10 CFR 50.67. The DBAs addressed in these attachments were selected from accidents that may involve damage to irradiated fuel. This guide does not address DBAs with radiological consequences based on technical specification reactor or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in this guide 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 NRC staff has determined that the analysis assumptions in the appendices to this guide provide an integrated approach to performing the individual analyses and generally expects licensees to address each assumption or propose acceptable alternatives. Such alternatives may be justifiable on the basis of plant-specific considerations, updated technical analyses, or, in some cases, a previously approved licensing basis consideration. The assumptions in the appendices are deemed consistent with the AST identified in Regulatory Position 3 and internally consistent with each other. Although licensees are free to propose alternatives to these assumptions for consideration by the NRC staff, licensees should avoid use of previously approved staff positions that would adversely affect this consistency. The NRC is committed to using probabilistic risk analysis (PRA) insights in its regulatory activities and will consider licensee proposals for changes in analysis assumptions based upon risk insights. The staff will not approve proposals that would reduce the defense in depth deemed necessary to provide adequate protection for public health and safety. In some cases, this defense in depth compensates for uncertainties in the PRA analyses and addresses	Complies The postulated accident radiological consequence analyses have been reanalyzed for AST (LAR 241, Enclosure 3, Section 1.1). For LOCA, see Appendix A For the FHA, see Appendix B For the MSLB, see Appendix E For the SGTR, see Appendix F For the LR, see Appendix G For the CRDE, see Appendix H The RVHD is not addressed in RG 1.183. It was also analyzed at the uprated conditions. No changes have been made to analysis assumptions based upon risk insights.

accident considerations not adequately addressed by the core damage frequency (CDF) and large early release frequency (LERF) surrogate indicators of overall risk.	
5.3 Meteorology Assumptions Atmospheric dispersion values (X/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining X/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 22, and 28). References 22 and 28 should be used if the FSAR X/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Ref. 29) implements Regulatory Guide 1.145 (Ref. 28) and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96 (Ref. 26) is generally acceptable to the NRC staff for use in determining control room X/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident X/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 30). All changes in X/Q analysis methodology should be reviewed by the NRC staff.	Complies The atmospheric dispersion (X/Q) values for the PBNP exclusion area boundary (EAB) and the low population zone (LPZ) are those from the current licensing basis. RG 1.194 guidance has been used for onsite X/Q values (LAR 241, Enclosure 3, Section 4.5). Fumigation has not been included since no credit is taken for an elevated release. ARCON96 was used for determining onsite X/Q values. Meteorological data acquired in accordance with the PBNP meteorological measurement program for the five-year period from 2000 to 2005 is used to calculate onsite atmospheric dispersion (LAR 241, Enclosure 3, Section 4.5).

6. ASSUMPTIONS FOR EVALUATING THE RADIATION DOSES FOR EQUIPMENT QUALIFICATION	
The assumptions in Appendix I to this guide are acceptable to the NRC staff for performing radiological assessments associated with equipment qualification. The assumptions in Appendix I will supersede Regulatory Positions 2.c(1) and 2.c(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 11), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in Appendix I, all other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective. *The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted.* 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. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs. TID14844) on EQ doses pending the outcome of the evaluation of the generic issue.	Complies Radiation environmental qualification of equipment analyses are not modified by this LAR. PBNP will continue to use the CLB qualification analyses, which are based on the TID-14844 source term (LAR 241, Enclosure 3, Section 1.4).

APPENDIX A

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LWR LOSS-OF-COOLANT ACCIDENT

1. Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Complies See Section 3, "Accident Source Term."
2. If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Complies Iodine chemical forms are consistent with the RG, i.e., 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide (LAR 241, Enclosure 3, Table 18). The sump pH is controlled at a value greater than 7.0 (LAR 241, Enclosure 3, Table 18).
ASSUMPTIONS ON TRANSPORT IN PRIMARY CONTAINMENT	
3. Acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the primary containment in PWRs or the drywell in BWRs are as follows: 3.1 The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	 Complies The activity initially released from the fuel is assumed to mix instantaneously and homogeneously throughout the assumed unsprayed volume of the containment (LAR 241, Enclosure 3, Section 6.1).

3.2 Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3). The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.	Complies For the containment leakage analysis, all activity released from the fuel is assumed to be in the containment atmosphere until removed by sprays, sedimentation, radioactive decay or leakage from the containment. Reduction of the airborne radioactivity in the containment by natural deposition is credited. The natural deposition removal coefficient for particulates was determined to be 0.1/hr. Instantaneous plate-out is not assumed (LAR 241, Enclosure 3, Section 6.1).
3.3 Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"[1] (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3). The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment	Complies Removal of elemental iodine from the containment atmosphere by spray is assumed to be terminated when the airborne inventory drops to 0.5 percent of the total elemental iodine released to the containment (this is a decontamination factor or DF of 200) (LAR 241, Enclosure 3, Section 6.1). The containment building is modeled as two discrete volumes, sprayed and unsprayed. The volumes are conservatively assumed to be mixed only by the containment fan coolers (LAR 241, Enclosure 3, Section 6.1). The particulate removal rate is reduced by a factor of 10 when a DF of 50 is reached (LAR 241, Enclosure 3, Section 6.1).

atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology). Footnote 1 - This document describes statistical formulations with differing levels of uncertainty. The removal rate constants selected for use in design basis calculations should be those that will maximize the dose consequences. For BWRs, the simplified model should be used only if the release from the core is not directed through the suppression pool. Iodine removal in the suppression pool affects the iodine species assumed by the model to be present initially.	
3.4 Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	N/A PBNP does not have post-accident in-containment air filtration systems. This position is not applicable to PBNP.

3.5 Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	N/A This position relates to suppression pool scrubbing in BWRs. The position is not applicable to PBNP.
3.6 Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).	N/A This position relates to activity retention in ice condensers. This position is not applicable to PBNP.
3.7 The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications. For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.	Complies PBNP is a PWR. A containment leak rate, based on the proposed technical specifications, of 0.2% per day of the containment air is assumed for the first 24 hours. After 24 hours, the containment leak rate is reduced to 0.1% per day (LAR 241, Enclosure 3, Table 18).

3.8 If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	N/A PBNP does not use the containment purge during power operation. During normal reactor operation at power, the containment may be vented by use of the containment gaseous and particulate sampling and monitoring penetrations. The system is automatically isolated in the event of a containment isolation signal.
ASSUMPTIONS ON DUAL CONTAINMENTS	
4. For facilities with dual containment systems, the acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the secondary containment or enclosure buildings are as follows.	N/A Regulatory Positions 4.1 through 4.6 apply to facilities with dual containment systems. These positions are not applicable to PBNP.
ASSUMPTIONS ON ESF SYSTEM LEAKAGE	
5. ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (Ref. A-7). The radiological consequences from the postulated leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of leakage from ESF components outside the primary containment for BWRs and PWRs.	Complies The radiological consequences from postulated ESF systems leakage is analyzed and combined with consequences postulated for containment leakage (LAR 241, Enclosure 3, Section 6.1).

5.1 With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the buildup of sump activity.	Complies Engineered Safety Feature (ESF) systems that recirculate water outside the primary containment (ECCS systems) are assumed to leak during their intended operation. Only iodine is released through this pathway since the noble gases are not assumed to dissolve in the sump and particulates would remain in the water of the ECCS leakage (LAR 241, Enclosure 3, Section 6.1).
5.2 The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	Complies Leakage from the ECCS system to the ESF rooms is 800 cc/min (two times the Program value of 400 cc/min). Recirculation is conservatively initiated at 0 minutes. 500 cc/min is assumed to flow to the RWST and 300 cc/min to the PAB. The assumption of the ECCS leakage beginning at 0 minutes is not consistent with the assumption of injection spray termination in the containment leakage portion of the analysis. However, beginning the ECCS leakage at 0 minutes adds conservatism to the dose consequences. The leakage continues for the 30-day period following the accident considered in the analysis (LAR 241, Enclosure 3, Section 6.1).
5.3 With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Complies With the exception of iodine, all radioactive materials in the recirculating liquid (sump water) are assumed to be retained in the liquid phase (LAR 241, Enclosure 3, Section 6.1).

5.4 If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: FF = (hf1 – hf2) / hfg Where: hf1 is the enthalpy of liquid at system design temperature and pressure; hf2 is the enthalpy of liquid at saturation conditions (14.7 psia, 212ºF); and hfg is the heat of vaporization at 212ºF.	Complies The temperature of the leakage was determined to exceed 212°F, and the flashing fraction was calculated based on the temperature of the containment sump liquid at the time recirculation begins.
5.5 If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Complies For the ECCS leakage to the auxiliary building, 10% of the total iodine in the leaked ECCS fluid is assumed to be available for release and is assumed to become airborne and leak directly to the environment from the initiation of recirculation through 30 days (LAR 241, Enclosure 3, Table 18).
5.6 The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Complies For ECCS leakage into the auxiliary building and RWST, the form of the released iodine is 97% elemental and 3% organic. No credit for holdup, filtration or dilution of ECCS leakage into the auxiliary building is taken (LAR 241, Enclosure 3, Table 18).

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

APPENDIX B ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT	
1. Source Term Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. 1.1 The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Complies It was assumed that a single fuel assembly is damaged, consistent with the CLB (LAR 241, Enclosure 3, Table 23).
1.2 The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Complies Consistent with PBNP CLB, all rods in one damaged fuel assembly are assumed to be breached. Does not comply It was assumed that the single damaged fuel assembly exceeds the RG 1.183 Footnote 11 limits. In lieu of the Table 3 Non-LOCA gap fractions, higher gap fractions were applied to the damaged assembly, following a method previously approved by the NRC for Kewaunee Power Station (ML070430020). The gap fractions used are those from Safety Guide 25 with the value for I-131 increased by 20%, to 0.12, consistent with the recommendation of NUREG/CR-5009 (LAR 241, Enclosure 3, Section 3.3).

1.3 The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. 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 is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Complies The chemical form of radioiodine released from the fuel to the spent fuel pool are assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide (LAR 241, Enclosure 3, Section 6.6).
2.0 Water Depth If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method.	Complies The minimum depth of water above the reactor vessel flange and above the top of irradiated assemblies during fuel movement is 23 feet (LAR 241, Enclosure 3, Table 23). The AST FHA dose analysis assumes an effective iodine DF of 200. This is equivalent to an elemental DF of 500 and an organic DF of 1 (LAR 241, Enclosure 3, Section 6.6).
3. Noble Gases The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Complies Noble gas is not retained in the pool water (LAR 241, Enclosure 3, Table 23). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (LAR 241, Enclosure 3, Section 6.6).

4. Fuel Handling Accidents Within the Fuel Building	N/A FHA in the Spent Fuel Pool was not analyzed. Since the assumptions and parameters used to model the release due to an FHA inside containment are identical to those for a FHA in the spent fuel pool, except for different CR intake atmospheric dispersion factors (X/Qs) for the different release paths, the activity released is the same regardless of the location of the accident. In order to bound the accident, the location with the highest X/Q value is assumed (LAR 241, Enclosure 3, Section 6.6, and Table 23).
5. Fuel Handling Accident Within Containment For fuel handling accidents postulated to occur within the containment, the following assumptions are acceptable to the NRC staff. 5.1 If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	N/A The containment is not isolated (LAR 241, Enclosure 3, Section 6.6).
5.2 If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	N/A The containment is open and not isolated following an FHA (LAR 241, Enclosure 3, Section 6.6).
5.3 If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.	Complies Activity is released over a 2-hour time period (LAR 241, Enclosure 3, Table 23).

5.4 A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	Complies No filtration is assumed (LAR 241, Enclosure 3, Table 23).
5.5 Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	N/A All activity released from the water is released from the containment to the environment over a 2-hour period. As such, the assumed containment volume is not credited with dilution or mixing (LAR 241, Enclosure 3, Table 23).

APPENDIX C ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BWR ROD DROP ACCIDENT
This appendix is not applicable to PBNP.

APPENDIX D ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BWR MAIN STEAM LINE BREAK ACCIDENT
This appendix is not applicable to PBNP.

APPENDIX E ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR MAIN STEAM LINE BREAK ACCIDENT	
This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a main steam line break accident at PWR light water reactors. These assumptions supplement the guidance provided in the main body of this guide[1]. Footnote 1 - Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," for acceptable assumptions and methodologies for performing radiological analyses.	
SOURCE TERMS	
1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.	Complies No fuel damage is postulated for the limiting event. The Source Term specifies only initial reactor coolant activity (LAR 241, Enclosure 3, Table 21).

2. If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed. Footnote 2 - The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DEI131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included. 2.1 A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).	Complies Two cases of iodine spiking are assumed (LAR 241, Enclosure 3, Section 6.4). Complies The pre-accident iodine spike case assumed that a reactor transient has occurred prior to the MSLB and has raised the RCS iodine concentration to a conservative value of 60 μCi/gm of DE I-131 (The TS 3.4.16 limit for a transient is 50 μCi/gm of DE I-131) (LAR 241, Enclosure 3, Section 6.4 and Table 21).
2.2 The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μ Ci/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8- hour spike exceeds that available for release from the fuel gap of all fuel pins.	Complies The accident-initiated iodine spike case, assumes that the primary system transient associated with the MSLB causes an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times the appearance rate corresponding to a maximum equilibrium RCS concentration of 0.5 μCi/gm of DE I-131 (proposed TS 3.4.16). The spike is allowed to continue until 4 hours from the start of the event. After this point in the accident there is no activity available for release from the gap (LAR 241, Enclosure 3, Section 3.4, Section 6.4 and Table 21).

3. The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Complies Accident-initiated iodine spike activity is instantaneously and homogeneously.mixed in the primary coolant. This assumption is inherent in the RADTRAD iodine spike model.
4. The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Complies Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic (LAR 241, Enclosure 3, Table 21).

TRANSPORT[3]	
5. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows. Footnote 3 - In this appendix, ruptured refers to the state of the steam generator in which primary-to-secondary leakage rate has increased to a value greater than technical specifications. Faulted refers to the state of the steam generator in which the secondary side has been depressurized by a MSLB such that protective system response (main steam line isolation, reactor trip, safety injection, etc.) has occurred. Partitioning Coefficient is defined as: *PC = mass of I per unit mass liquid / mass of I per unit mass gas*	
5.1 For facilities that have not implemented alternative repair criteria (see Ref. E-1, DG-1074), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Complies Note that the primary-to-secondary leak rate is specified per SG, rather than per plant. Thus, leakage is apportioned according to the per SG leakage limit (LAR 241, Enclosure 3, Section 2.2).
5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g.,lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft3).	Complies Appropriate density is used to insure that the accident-induced leak rate is greater than the operational leak rate (LAR 241, Enclosure 3, Section 2.2).

5.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Complies The primary-to-secondary leak rate to the affected SG is assumed to continue until the temperature of the leakage is less than 212°F, which is assumed at 60 hours. The release of radioactivity from the unaffected SG continues for 30 hours (time to place RHR in operation) (LAR 241, Enclosure 3, Table 21).
5.4 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Complies All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere (LAR 241, Enclosure 3, Section 6.4).

5.5 The transport model described in this section should be utilized for iodine and particulate releases from the steam generators. This model is shown in Figure E-1 and summarized below. 5.5.1 A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. During periods of steam generator dryout, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence.	Complies The affected SG is assumed to boil dry (iodine partition = 1.0) (LAR 241, Enclosure 3, Table 21). Primary-to-secondary leakage to the unaffected SG is assumed to mix with the secondary water without flashing. The partition factor in the unaffected SG = 0.01 (LAR 241, Enclosure 3, Table 21).
5.5.2 The leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" (Ref. E-2), during periods of total submergence of the tubes.	Complies Flashing leakage is not assumed to the unaffected SG. The affected SG is assumed to boil dry. Thus, no credit is taken for scrubbing of primary activity from flashed rupture flow (LAR 241, Enclosure 3, Section 6.4). Also, LAR 241, Enclosure 3, Table 21 does not specify flashing leakage or scrubbing credit.
5.5.3 The leakage that does not immediately flash is assumed to mix with the bulk water.	Complies Leakage to the unaffected SG is assumed to mix with the bulk SG water (LAR 241, Enclosure 3, Section 6.4).

5.5.4 The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.	Complies Analysis assumes a partition factor of 0.01. See response to Regulatory Position 5.5.1 of this appendix. A particulate retention factor of 0.0025 is applied (LAR 241, Enclosure 3, Table 21).
5.6 Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. E-3). The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.	Complies Steam generator tube bundle uncovery is not predicted or postulated (unaffected SG discussion for SGTR also applies to MSLB) (LAR 241, Enclosure 3, Section 6.2). Water level in the unaffected SG was specifically analyzed for the SGTR.

APPENDIX F ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR STEAM GENERATOR TUBE RUPTURE ACCIDENT	
This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a steam generator tube rupture accident at PWR light-water reactors. These assumptions supplement the guidance provided in the main body of this guide.1	
SOURCE TERM	
1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Complies No fuel damage is postulated to occur for the SGTR event. The Source Term specifies only initial reactor coolant activity (LAR 241, Enclosure 3, Table 19).
2. If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification. Two cases of iodine spiking should be assumed. Footnote 2 - The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DEI131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	Complies No fuel damage is postulated to occur for the SGTR event. Two cases of iodine spiking are assumed (LAR 241, Table 19).

2.1 A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μ Ci/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).	Complies Case assumes a reactor transient prior to the postulated SGTR that raises the RCS iodine concentration to a conservative value of 60 μCi/gm of dose equivalent (DE) I-131. (The Technical Specification (TS 3.4.16) limit for a transient is 50 μCi/gm of dose equivalent (DE) I-131.) This is the pre-accident spike case (LAR 241, Enclosure 3, Table 19).
2.2 The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μ Ci/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8- hour spike exceeds that available for release from the fuel gap of all fuel pins.	Complies The accident-initiated iodine spike case, assumes that the primary system transient associated with the MSLB causes an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 335 times the appearance rate corresponding to a maximum equilibrium RCS concentration of 0.5 μCi/gm of DE I-131 (proposed TS 3.4.16). The spike is allowed to continue until 8 hours from the start of the event (LAR 241, Enclosure 3, Table 19).
3. The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.	Complies The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant. This assumption is inherent in the RADTRAD model.
4. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Complies Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic (LAR 241, Enclosure 3, Table 19).

TRANSPORT[3]	
Footnote 3 - In this appendix, ruptured refers to the state of the steam generator in which primary-to-secondary leakage rate has increased to a value greater than technical specifications.	
5. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows: 5.1 The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Complies The primary-to-secondary leak rate is specified per SG, rather than per plant. Thus, leakage is apportioned according to the per SG leakage limit (LAR 241, Enclosure 3, Section 2.2).
5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft3).	Complies The analysis leak rate is significantly greater than the operational leak rate (LAR 241, Enclosure 3, Section 2.2).
5.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100° C (212° F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Complies The release of radioactivity from the affected SG is assumed to continue for 30 minutes. The release of radioactivity from the unaffected SG is assumed to continue until shutdown cooling is in operation and the steam release from the SGs is terminated, 30 hours into the event (LAR 241, Enclosure 3, Table 19).

5.4 The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Complies The release of fission products from the secondary system is evaluated with the assumption of a coincident loss of offsite power (LOOP) (LAR 241, Enclosure 3, Table 19).
5.5 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Complies All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere (LAR 241, Enclosure 3, Section 6.2).
5.6 The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	
Appendix E, Regulatory Position 5.5.1: A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. • During periods of steam generator dryout, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. • With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence.	Complies Pre-trip and post-trip flashing fractions were calculated (LAR 241, Enclosure 3, Section 6.2). Flashing leakage is not assumed to the unaffected SG (LAR 241, Enclosure 3, Table 19). Table 19 does not cite flashing leakage to the unaffected SG. All analyses with unaffected SGs use the same unaffected SG transport model. N/A Sufficient water level is maintained in both SGs (LAR 241, Enclosure 3, Section 6.2).

Appendix E, Regulatory Position 5.5.2: The leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" (Ref. E-2), during periods of total submergence of the tubes.	Complies Flashing rupture flow is assumed to the affected SG. No credit is taken for scrubbing of primary activity from flashed rupture flow (LAR 241, Enclosure 3, Section 6.2).
Appendix E, Regulatory Position 5.5.3: The leakage that does not immediately flash is assumed to mix with the bulk water.	Complies The fraction of primary coolant iodine that is not assumed to become airborne immediately, mixes with the secondary water, and is assumed to become airborne at a rate proportional to the steaming rate (LAR 241, Enclosure 3, Section 6.2).
Appendix E, Regulatory Position 5.5.4: The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.	Complies Analysis assumes a partition factor of 0.01. A particulate retention factor of 0.0025 is applied (LAR 241, Enclosure 3, Table 19).
Appendix E, Regulatory Position 5.6: Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. E-3). The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.	Complies Steam generator tube bundle uncovery is not predicted or postulated (LAR 241, Enclosure 3, Section 6.2). Water level in the unaffected SG was specifically analyzed for the SGTR.

APPENDIX G ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR LOCKED ROTOR ACCIDENT	
This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a locked rotor accident at PWR light water reactors.1 These assumptions supplement the guidance provided in the main body of this guide.	
SOURCE TERM	
1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Complies Analysis assumed 30% of the fuel rods in core are breached (LAR 241, Enclosure 3, Table 20).
2. If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment.	Complies See Regulatory Position 1 above.
3. The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Complies The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant. Instantaneous, homogeneous mixing is inherent in the RADTRAD model.
4. The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Complies Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic (LAR 241, Enclosure 3, Table 20).

RELEASE TRANSPORT	
5. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows. 5.1 The primary-to-secondary leak rate in the steam generators should be assumed to be the leak-rate-limiting condition for operation specified in the technical specifications. The leakage should be apportioned between the steam generators in such a manner that the calculated dose is maximized.	Complies The primary-to-secondary leak rate is specified per SG, rather than per plant. Thus, leakage is apportioned according to the per SG leakage limit (LAR 241, Enclosure 3, Section 2.2).
5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft3).	Complies Appropriate density is used to insure that the accident-induced leak rate is greater than the operational leak rate (LAR 241, Enclosure 3, Section 2.2).
5.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100° C (212° F). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Complies The primary-to-secondary leakage is assumed to continue until shutdown cooling is in operation and the steam release from the SGs is terminated (30 hours into the event) (LAR 241, Enclosure 3, Section 6.3).
5.4 The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Complies The release of fission products from the secondary system is evaluated with the assumption of a coincident loss of offsite power (LOOP) (LAR 241, Enclosure 3, Section 1.4).

5.5 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Complies All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere (LAR 241, Enclosure 3, Section 6.3).
5.6 The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Complies The SG activity transport and release model for the LR is consistent with the unaffected SG assumptions of the MSLB. See Appendix F of this document.

APPENDIX H ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR ROD EJECTION ACCIDENT	
This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a rod ejection accident at PWR light water reactors[1]. These assumptions supplement the guidance provided in the main body of this guide. Footnote 1 - Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity" (USNRC, December 1998), for acceptable assumptions and methodologies for performing radiological analyses.	
SOURCE TERM	
1. Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.	Complies The fraction of activity released from melted fuel to containment is 50% for iodine. This is conservative with respect to RG 1.183, Appendix H, Position 1 which states that for the containment leakage release path model, the available inventory from the melted fuel is 25% for iodine (LAR 241, Enclosure 3, Section 6.5).
2. If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the loss-of-coolant accident (LOCA), main steam line break, and steam generator tube rupture.	Complies Fuel damage is postulated. See Position 1, above.

3. Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.	Complies Two release cases are considered (LAR 241, Enclosure 3, Section 6.5). The assumption of instantaneous and homogeneous mixing of activity released from the fuel into the containment atmosphere and primary coolant is inherent in the RADTRAD model.
4. The chemical form of radioiodine released to the containment atmosphere should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. If containment sprays do not actuate or are terminated prior to accumulating sump water, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case basis. Evaluations of pH should consider the effect of acids created during the rod ejection accident event, e.g., pyrolysis and radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Complies The chemical form of radioiodine released from the damaged fuel to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide (LAR 241, Enclosure 3, Table 22). Containment sump pH is controlled to a value of 7 or greater (LAR 241, Enclosure 3, Table 18).
5. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Complies The chemical form of radioiodine released from the SGs to the environment is assumed to be 97% elemental iodine, and 3% organic iodide (LAR 41, Enclosure 3, Table 22).

TRANSPORT FROM CONTAINMENT	
6. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the containment are as follows. 6.1 A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to Appendix A to this guide for guidance on acceptable methods and assumptions for evaluating these mechanisms.	Complies For the containment leakage case, sedimentation of alkali metal particulates in containment is credited. Containment spray is not credited (LAR 241, Enclosure 3, Table 22).
6.2 The containment should be assumed to leak at the leak rate incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing. Leakage from subatmospheric containments is assumed to be terminated when the containment is brought to a subatmospheric condition as defined in technical specifications.	Complies The containment is assumed to leak at the proposed TS maximum allowable rate of 0.2% for the first 24 hours and 0.1% for the remainder of the event (LAR 241, Enclosure 3, Table 22).

TRANSPORT FROM THE SECONDARY SYSTEM	
7. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the secondary system are as follows. 7.1 A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.	Complies The assumed accident-induced leak rate is greater than the operational leak rate (LAR 241, Enclosure 3, Section 2.2). Does not comply Primary-to-secondary leakage is assumed to continue for 2000 seconds. This time is based on the time to equalize primary and secondary pressures, for a 2" LOCA (LAR 241, Enclosure 3, Section 6.5). Steam releases are assumed to continue for 30 hours (LAR 241, Enclosure 3, Table 22).
7.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft3).	Complies Appropriate density is used to insure that the accident-induced leak rate is greater than the operational leak rate (LAR 241, Enclosure 3, Section 2.2).
7.3 All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.	Complies All of the noble gas released to the secondary side is assumed to be released directly to the environment (LAR 241, Enclosure 3, Section 6.5).

7.4 The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Complies The SG activity transport and release model for the CRDE is consistent with the unaffected SG assumptions of the MSLB and SGTR. See Appendix F of this document.

APPENDIX I ASSUMPTIONS FOR EVALUATING RADIATION DOSES FOR EQUIPMENT QUALIFICATION	
This appendix addresses assumptions associated with equipment qualification that are acceptable to the NRC staff for performing radiological assessments. As stated in Regulatory Position 6 of this guide, this appendix supersedes Regulatory Positions 2.c.(1) and 2.c.(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (USNRC, June 1984), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in this appendix, other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective.	N/A Radiation environmental qualification of Equipment analyses are not modified by this LAR. PBNP will continue to use the CLB qualification analyses, which are based on the TID-14844 source term (LAR 241, Enclosure 1, Section 5.4). The NRC Staff concluded that there was no clear basis for backfitting the requirement to modify the design basis for equipment qualification to adopt the AST. Post-accident vital access shielding is addressed in LAR 241, Enclosure 1, Section 5.6.