10 CFR 50.67
10 CFR 50.90

CP&L
A Progress Energy Company

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United States Nuclear Regulatory Commission
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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING
FULL IMPLEMENTATION OF ALTERNATIVE RADIOLOGICAL SOURCE TERM

Ladies and Gentlemen:

In accordance with the provisions of the Code of Federal Regulations, Title 10 (10 CFR), Part 50.67 and Part 50.90, Carolina Power & Light (CP&L) Company requests review and approval of full implementation of an alternative source term (AST) for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. CP&L also requests an amendment to the HBRSEP, Unit No. 2, Technical Specifications (TS) which implements various input assumptions in those analyses that were revised to implement the AST. The reanalysis of the dose consequences of accidents discussed in this submittal is necessary to support operation at an uprated reactor core power level bounded by a core power level of 2346 megawatts thermal. This submittal does not request an uprate of reactor core power level. CP&L intends to submit a separate request for review and approval of the proposed reactor core power uprate. A request for a selective implementation of an AST in the HBRSEP, Unit No. 2, Fuel Handling Accident analysis was previously submitted by letter dated March 13, 2002.

The proposed TS changes would revise the definition of Dose Equivalent I-131, revise the requirements associated with primary to secondary leakage through the steam generators, revise the requirements associated with Reactor Coolant System specific activity, and revise the description of the Explosive Gas and Storage Tank Radioactivity Monitoring Program. The proposed changes are necessary to provide assurance that input assumptions to the reanalyses are adequately protected.

Attachment I provides an affirmation as required by 10 CFR 50.30(b).

Attachment II provides a description of the current condition, a description of the proposed change, and a safety assessment of the proposed change.

Attachments III and IV provide annotated TS pages showing the proposed changes, and the retyped TS pages incorporating the proposed changes, respectively.

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Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

Attachment V provides the atmospheric dispersion factors used in the affected analyses.

Attachment VI provides the fission product inventories used in the affected analyses.

Attachment VII provides the ventilation system parameters used in the affected analyses

Attachments VIII through XIII provide the key parameter inputs used in the analysis of the dose consequences of the events discussed in Attachment II. These Attachments also provide the does consequence results of the analyses.

Attachment XIV provides an annotated site drawing that highlights the release and receptor points assumed in the analyses.

Enclosure (1) is a floppy disk containing the input files for calculation of the Onsite Atmospheric Dispersion Factors in ARCON96 format.

In accordance with 10 CFR 50.91(b), CP&L is providing the State of South Carolina with a copy of the proposed license amendment.

CP&L requests approval of the proposed License Amendment by September 1, 2002, with the amendment being implemented within 60 days of approval. The approval date was selected to support planning activities for Refueling Outage (RO)-21.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom.

Sincerely,

B. L. Fletcher III
Manager - Regulatory Affairs

CWS/cws

Attachments:

I. Affirmation
II. Request For Technical Specifications Change Regarding Full Implementation of Alternative Radiological Source Term
III. Markup of Technical Specifications Pages
IV. Retyped Technical Specifications Pages
V. Atmospheric Dispersion Factors
VI. Fission Product Inventory
VII. Ventilation System Parameters
VIII. Main Steam Line Break (MSLB)
IX. Reactor Coolant Pump Shaft Seizure (Locked Rotor)
X. Single Rod Control Cluster Assembly (RCCA) Withdrawal
XI. Steam Generator Tube Rupture (SGTR)
XII. Large Break Loss-of-Coolant Accident (LBLOCA)
XIII. Waste Gas Decay Tank (WGDT) Rupture
XIV. Annotated Site Drawing

Enclosure:

(1) Electronic media containing ARCON96 input files, output files, and Joint Frequency Distribution Tables

c: (w/o enclosure)

Mr. L. A. Reyes, NRC, Region II
Mr. H. J. Porter, Director, Division of Radioactive Waste Management (SC)
Mr. R. M. Gandy, Division of Radioactive Waste Management (SC)
Mr. R. Subbaratnam, NRC, NRR
NRC Resident Inspector, HBRSEP
Attorney General (SC)

AFFIRMATION

The information contained in letter RNP-RA/02-0067 is true and correct to the best of my information, knowledge and belief; and the sources of my information are officers, employees, contractors, and agents of Carolina Power & Light Company. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 10th Day of May 2002

J. W. Moyer
Vice President, HBRSEP, Unit No. 2

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING
FULL IMPLEMENTATION OF ALTERNATIVE RADIOLOGICAL SOURCE TERM

DESCRIPTION OF CURRENT CONDITION

The current H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, licensing basis for the radiological consequences analyses for accidents discussed in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR) is based on methodologies and assumptions that are derived from Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactors," 1962.

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, which provides a mechanism for licensed power reactors to replace the traditional accident source term used in the design basis accident analyses with alternative source terms (ASTs). Part 50.67 requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of the affected design basis accidents. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." A request for selective implementation of an AST in the HBRSEP, Unit No. 2, Fuel Handling Accident analysis was previously submitted by letter dated March 13, 2002.

DESCRIPTION OF THE PROPOSED CHANGE

Carolina Power and Light (CP&L) Company proposes to revise the HBRSEP, Unit No. 2, licensing basis to implement the AST, described in RG 1.183, through reanalysis of the radiological consequences of the UFSAR Chapter 15 accidents discussed in this submittal. As part of the full implementation of this AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11. The following UFSAR Chapter 15 accident dose consequences are discussed in this submittal:

- Main Steam Line Break,
- Reactor Coolant Pump Shaft Seizure (Locked Rotor),
- Single Rod Control Cluster Assembly (RCCA) Withdrawal,
- Steam Generator Tube Rupture (SGTR),
- Large Break Loss-of-Coolant Accident (LBLOCA), and
- Waste Gas Decay Tank (WGDT) Rupture.

Although the dose consequences of the WGDT Rupture event are not dependent upon reactor power or the reactor core source term, the analysis is updated to incorporate new atmospheric dispersion factors and to evaluate the dose consequences to the revised TEDE criteria.

The following changes to the HBRSEP, Unit No. 2, Technical Specifications (TS) are proposed:

> The definition of Dose Equivalent I-131 in Section 1.1 is revised to reference NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation," December 1997, as the source of thyroid dose conversion factors.
>
> The Reactor Coolant System (RCS) operational leakage limits, stated in Limiting Condition for Operation (LCO) 3.4.13, "RCS Operational Leakage," for total primary to secondary leakage through all steam generators is reduced from 1 gpm to 0.3 gpm. In addition, the limit specified for primary to secondary leakage through any one steam generator is reduced from 500 gallons per day to 150 gallons per day.
>
> The Dose Equivalent I-131 requirements of TS 3.4.16, "RCS Specific Activity," are reduced from 1.0 μCi/gm to 0.25 μCi/gm in Condition A and in Surveillance Requirement (SR) 3.4.16.2. In addition, Figure 3.4.16-1, "Reactor Coolant Dose Equivalent I-131 Specific Activity Limit Versus Percent of Rated Thermal Power," is deleted. Required Action A.1 is revised to replace the reference to the acceptable region of Figure 3.4.16-1 with a limit of ≤ 60.0 μCi/gm. The second entry condition of Condition C is revised to replace the reference to the unacceptable region of Figure 3.4.16-1 with reference to > 60 μCi/gm.
>
> The description of the Explosive Gas and Storage Tank Radioactivity Monitoring Program provided in TS 5.5.12 is revised to incorporate the TEDE as the acceptance criteria for dose consequences.

SAFETY ASSESSMENT

Atmospheric Dispersion (X/Q) Factors

The historical X/Q data provided in the HBRSEP, Unit No. 2, UFSAR did not meet CP&L's expectations for level of detail for documentation and design bases, and was therefore, not considered sufficient for the revision of the limiting design basis analyses using the AST methodology. New X/Q factors were required to be calculated for release-receptor pairs that were not previously evaluated. Therefore, new X/Q factors for the Control Room (CR) and Technical Support Center/Emergency Operations Facility (TSC/EOF) were calculated to place all of these design input X/Q factors on a common, consistent calculational basis. In addition, plant-specific X/Q factors for offsite dose consequences were developed. Meteorological data over a nine year period (1988 through 1996) was used in the development of the new X/Q factors.

For offsite receptor locations, the new χ/Q factors were developed using the PAVAN computer code ("PAVAN: An Atmospheric Dispersion Program for Evaluating Design Bases Accident Releases of Radioactive Materials from Nuclear Power Stations," NUREG/CR-2858, November 1982. RSICC Computer Code Collection No. CCC-445). The offsite χ/Q factors are provided in Attachment V, Table 1, "Offsite Atmospheric Dispersion Factors (χ/Q)."

New χ/Q factors for onsite release-receptor combinations were developed using the ARCON96 computer code ("Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Rev. 1, May 1997. RSICC Computer Code Collection No. CCC-664). New guidance, which supersedes the NUREG/CR-6331 recommendations for using certain default parameters as input, contained in NRC Draft Regulatory Guide DG-1111, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," December 2001, has been implemented. Accordingly, the following changes from the default values were made:

1. Surface roughness length, m: A value of 0.2 is used in lieu of the default value of 0.1, and

2. Averaging sector width constant: A value of 4.3 is used in lieu of the default value of 4.0.

New χ/Q factors have been determined for the following release-receptor location combinations:

- Plant stack to Control Room intake,
- Closest Main Steam Safety Valve (MSSV)/Power Operated Relief Valve (PORV) to Control Room intake,
- Closest main steam line to Control Room,
- Closest point from containment to Control Room intake,
- Closest point from containment to TSC/EOF intake,
- Residual heat removal (RHR) heat exchanger room to Control Room intake,
- RHR heat exchanger room to TSC/EOF intake, and
- Fuel Handling Building (FHB) wall to Control Room intake.

Attachment V, Table 2, "Release-Receptor Combination Parameters," provides information related to the relative elevations of the release-receptor combinations, the straight-line horizontal distance between the release point and the receptor location, and the direction (azimuth) from the receptor location to the release point. HBRSEP, Unit No. 2, is located between the 7° West and the 6° West Declination Line at approximately 6°23'. As a result, "plant north" must be adjusted to "true north" by the addition of 6.38°. Attachment XIV provides an enlargement of a section of UFSAR, Figure 1.2.2-1, "Plot Plan," that has been annotated to highlight the release and receptor point locations described above. All releases are taken as ground releases in accordance with the guidance provided in Nuclear Energy Institute (NEI), "Control Room Habitability Assessment Guidance," NEI 99-03, June 2001. Attachment V, Table 3, "Onsite Atmospheric Dispersion Factors (χ/Q)," provides the Control Room χ/Q factors for the release-receptor combinations listed above. These factors are not corrected for occupancy. Attachment V, Table 4, "Onsite Atmospheric Dispersion Factors (χ/Q)," provides the Control Room χ/Q factors corrected for occupancy using the guidance provided in RG 1.183, Section 4.2.6.

Discussions of the HBRSEP, Unit No. 2, meteorology system and data collection were previously submitted in the HBRSEP, Unit No. 2, letter dated March 13, 2002, requesting selective implementation of an AST in the Fuel Handling Accident analysis. The March 13, 2002, letter also provided the nine years of meteorological data, the PAVAN input and output files for the offsite χ/Q factors, as well as the ARCON96 files associated with the Fuel Handling Accident analysis of the Control Room dose consequences on electronic media. Therefore, those discussions and data are not repeated in this submittal.

Fission Product Inventory

The HBRSEP, Unit No. 2, reactor core consists of 157 fuel assemblies, with each fuel assembly containing 204 fuel rods. The full core isotopic inventory was determined in accordance with RG 1.183, Regulatory Position 3.1. The plant-specific isotopic source terms were developed using a bounding approach. RG 1.183, Regulatory Position 3.1, recommends that the core isotopic inventory be determined using an appropriate isotope generation and depletion computer code, such as ORIGIN-2 or ORIGEN-ARP. HBRSEP, Unit No. 2, used the ORIGEN-S computer code (part of the SCALE-4.3 system of codes) to develop the isotopics for a variety of burnup, enrichment, and burnup rates (power levels). The ORIGEN-S code is used for the following reasons:

1. ORIGEN-2 has only a two cross-section set to represent all Pressurized Water Reactor (PWR) fuel, and basically all enrichments. ORIGEN-S has a cross-section set tailored to 15x15 fuel and is specifically designed to address the range of expected burnups and initial enrichments expected at HBRSEP, Unit No. 2. The input data for this 15x15 fuel cross section development was reviewed and the pin geometries for Westinghouse and Exxon fuel (formerly Siemens and now Framatome ANP) are nearly identical. Therefore, these cross-sections were considered suitable for use.
2. ORIGEN-S has seen the highest level of recent support and benchmarking as part of the U.S. Department of Energy, Office of Civilian Radioactive Waste Management High Level Waste Program.
3. Tests (not included here) indicate that ORIGEN-S and ORIGEN-2 are in reasonable agreement. Both programs are more conservative than TID-14844 for the most important I-131 isotope.
4. The NRC is a significant sponsor of the SCALE-4.3 package development.

Sensitivity studies were run with various combinations of burnups and enrichments to identify a bounding single assembly isotopic source term. A bounding burnup of 60,000 MWD/MTU and a bounding enrichment of 4.95 w/o (up to 5.0 w/o bounded by sensitivity study), were used. For rod average burnups in excess of 54,000 MWD/MTU, the heat generation rate was limited to 6.12 kw/ft. The assembly source term was multiplied by 1.02 to reflect operation prior to shutdown at 102% of rated power (102% of 2300 MWt or 2346 MWt). A bounding radial peaking factor of 1.8 was then applied to simulate the effect of conservative application of local peaking factors which might affect localized fuel failures for assemblies containing the peak fission product inventory. Attachment VI, Table 1, provides the 60 isotopes (noble gas, halogen, and alkali metal) required by RG 1.183; NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation," December 1997, and Supplement 1, June 8, 1999; and NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995, for the Loss-of-Coolant Accident (LOCA) dose consequences analysis. Attachment VI, Table 2, provides those isotopes considered for the non-LOCA events.

The core inventory release fractions for the gap release and early in-vessel damage phases for the design basis LOCAs and the Single RCCA Withdrawal accident utilized those release fractions provided in RG 1.183, Regulatory Position 3.2, Table 2, "PWR Core Inventory Fraction Released Into Containment." For non-LOCA events, the fractions of the core inventory assumed to be in the gap are consistent with RG 1.183, Regulatory Position 3.2, Table 3, "Non-LOCA Fraction of Fission Product Inventory in Gap."

Secondary coolant system activity is limited to a value of ≤ 0.1 μCi/gm in accordance with TS 3.7.15. Noble gases entering the secondary coolant system are assumed to be immediately released; thus, the noble gas activity concentration in the secondary coolant system is assumed to be 0.0 μCi/gm. RG 1.183 does not address secondary coolant equilibrium specific activity for the alkali metals. The equilibrium specific activity of the alkali metals resulting from steam generator (SG) leakage from the RCS is assumed to be 10% of the primary coolant equilibrium concentration.

Ventilation Systems

Control Room Ventilation System Description:

The Control Room air conditioning system consists of an environmental control system and an air cleanup system to serve the Control Room, as described in the UFSAR, Sections 9.4.2 and 6.4. The environmental control system operates continually during normal and emergency operation. This system consists of redundant 100% capacity centrifugal fans and gravity dampers arranged in parallel and a stainless steel housing containing a medium efficiency filter and redundant 100% capacity direct expansion cooling coils. Redundant 100% capacity service water-cooled condensing units are provided, each connected by refrigerant piping to each cooling coil. The air cleanup system normally operates only during emergency conditions. This system consists of redundant 100% capacity centrifugal fans and gravity dampers arranged in parallel and a stainless steel housing containing a pre-filter, a pre-high efficiency particulate adsorber (HEPA), charcoal adsorber, and post-HEPA filter banks. The Control Room air conditioning system contains a single outside air intake with connecting duct containing redundant air operated control dampers in parallel. This outside air intake is located on the east wall of the Control Room through louver number L-19. A diagram of the air conditioning system is provided in the UFSAR, Figure 9.4.1-4.

The Control Room air conditioning system is designed to provide three operational modes: normal ventilation, emergency pressurization, and emergency recirculation. During normal ventilation, one train of the environmental control system is in operation. During emergency pressurization, a single train of both the environmental control system and the air cleaning system are in operation. A positive pressure is maintained in the Control Room envelope with respect to the outdoors and adjacent areas, with one exception. This exception occurs when there is a loss of Auxiliary Building exhaust fan HVE-7 concurrent with Control Room pressurization. In this case, testing has shown that the Hagan Room could be slightly positive with respect to the Control Room. However, procedures require actions to be taken within one hour to assure that Hagan Room pressure is reduced. A Safety Injection (SI) signal or a signal from the Control Room area radiation monitor will automatically place the system in the emergency pressurization mode. The emergency pressurization mode may also be manually selected.

The Emergency Recirculation mode of operation is activated by first placing the system in the emergency pressurization mode and then closing both outside air intake dampers via their control switches in the Control Room. This mode of operation is not a design basis requirement, but is provided to allow flexibility to isolate the Control Room outside air makeup.

Control Room Unfiltered Air Inleakage:

The current Control Room dose analyses assume an unfiltered air inleakage of 15 cfm from unidentified sources and 70 cfm from the Hagan Room (a total unfiltered inleakage flow rate of 85 cfm). The value assumed in the AST analyses (except for the LOCA analysis) for Control Room unfiltered air inleakage is 300 cfm from all sources based on a qualitative evaluation. Consideration was given to the magnitudes of the known performance parameters of the Control Room ventilation system. For example, the filtered makeup air flow during pressurization is known to be limited to 400 cfm. Other sites that have measured unfiltered air inleakage (using currently acceptable methods) have generally measured less than 300 cfm from all sources. Outliers exist to this general experience, such as Control Rooms opening to much larger structures or buildings, but 300 cfm is considered to be a conservative assumption for the HBRSEP, Unit No. 2, Control Room. The LOCA analysis assumes an unfiltered air inleakage value of 170 cfm from all sources. When the 70 cfm Hagan Room inleakage is removed after one hour, this results in an unfiltered inleakage rate of 100 cfm from unknown sources. This inleakage rate is slightly greater than six times the unfiltered inleakage rate from unknown sources that is currently assumed in the LOCA analysis, i.e., 15 cfm. This is considered to be a more conservative assumption, resulting in a more conservative design and licensing basis for the Control Room habitability envelope.

For those events that do not result in the initiation of an SI signal (Locked Rotor and Single RCCA Withdrawal), the associated dose consequence analyses assume one hour for the switch from the normal to emergency pressurization mode (except for the SGTR event, which assumes 310 seconds). These analyses assume that outside air is drawn into the Control Room at the normal ventilation flow rate of 400 cfm in addition to the 300 cfm unfiltered inleakage during the first hour of the event. The switch in ventilation mode is initiated either by an automatic signal from the Control Room area radiation monitor or by manual operator action. One hour for manual operator action is considered to be a conservative assumption because the controls required to place the Control Room ventilation system in the emergency pressurization mode are located within the Control Room and procedural guidance for switching to the emergency pressurization mode is provided to the operators. It is expected that actions required to place the Control Room ventilation system in the emergency pressurization mode can be accomplished in a significantly shorter period of time. Attachment VII, Table 1, "Control Room Ventilation System Parameters," provides those key Control Room ventilation parameters that are assumed in the analyses discussed in this submittal.

CP&L is aware of ongoing discussions between the NRC and the Industry concerning Control Room habitability issues. In recognition of this, CP&L commits to perform a leak rate test on the HBRSEP, Unit No. 2, Control Room envelope prior to implementation of the changes requested in this submittal. Upon completion of this testing, CP&L will provide the results in a supplement to this submittal. In addition to the results of the testing, CP&L will also consider the following points in relation to this testing:

- A single value for unfiltered Control Room air inleakage will be established with a basis (tracer gas testing),
- In the event that the analyses contained in this submittal do not bound the new established value, the applicable analyses will be revised to bound the new established value,
- The testing and reanalysis, if required, is intended to demonstrate compliance with the Control Room dose acceptance criteria of 10 CFR 50, Appendix A, GDC-19. In the event compliance with the GDC-19 dose acceptance criteria can not be supported by the current licensing basis, a comprehensive corrective action plan to restore compliance with the GDC-19 dose acceptance criterion will be developed.

TSC/EOF Ventilation System:

The TSC/EOF ventilation systems are described in the HBRSEP, Unit No. 2, UFSAR, Section 9.4.11. The TSC/EOF Building is supported by a normal ventilation system and an emergency air filtration (EAF) system. The EAF system is a once-through charcoal and HEPA filtration system designed to pressurize and filter the ventilation intake air to reduce the level of contaminates in the intake air. The EAF system may be started manually, or automatically upon an actuation signal from the TSC/EOF Building Ventilation Monitor. Upon actuation, the intake air is redirected from the normal air handling units to the EAF units. The filtered air is then supplied to the normal air handling units. The filtered air drawn from the outside pressurizes the TSC/EOF, relative to the outside atmosphere, which prevents the infiltration of unfiltered outside air. The analysis for the LBLOCA assumes an unfiltered inleakage rate of 500 cfm into the EOF/TSC envelope for the duration of the event. No reduction of dose is assumed to occur due to recirculation filtration because the TSC/EOF ventilation system does not recirculate the air.

Main Steam Line Break (MSLB)

Background:

This event is caused by a double-ended break of one main steam line outside containment with the reactor operating at hot full power. The radiological consequences of such an accident bound those of a MSLB inside containment. Upon a MSLB, the affected SG rapidly depressurizes and releases the initial contents of the SG to the environment. The rapid secondary depressurization causes a reactor power transient, resulting in a reactor trip. Plant cooldown is achieved via the remaining unaffected SGs. This event is described in the UFSAR, Section 15.1.5.

Compliance with RG 1.183 Regulatory Positions:

The revised MSLB dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix E, "Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident," as discussed below:

1. Regulatory Position 1 - The core inventory of the radionuclide groups utilized for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided as Attachment VI, Table 2. The fraction of fission product inventory in the gap available for release due to fuel breach is consistent with Table 3 of RG 1.183.
2. Regulatory Position 2 - No fuel damage is postulated for the current analysis as stated in the UFSAR, Section 15.6.4. However, an additional and bounding MSLB event is analyzed, assuming the breach failure of two fuel assemblies.
3. Regulatory Position 2.1 - A case is analyzed assuming a reactor transient prior to the postulated MSLB (i.e., a pre-accident iodine spike case). This case assumes that the pre-MSLB primary coolant iodine concentration has been raised to 60 μCi/gm Dose Equivalent (DE) I-131.
4. Regulatory Position 2.2 - A case is analyzed assuming that the MSLB causes an iodine spike in the primary system. This concurrent iodine spike case assumes that the activity from the fuel to the reactor coolant is released at a rate of 500 times the iodine equilibrium release rate consistent with two times the proposed TS 3.4.16 limit of 0.25 μCi/gm DE I-131.
5. Regulatory Position 3 - The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.
6. Regulatory Position 4 – The chemical form of radioiodine released from the fuel was assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the faulted SG and the unaffected SGs to the environment are assumed to be 97% elemental and 3% organic. These fractions apply as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

7. Regulatory Position 5.1 - The primary to secondary leak rate is apportioned between the SGs in such a manner that the calculated dose is maximized. The proposed TS 3.4.13 requirements limit leakage through all SGs to 0.3 gpm and leakage through any one SG to 150 gallons per day. Since the tube leak into the faulted SG and subsequently to the environment continues until reactor coolant temperature drops below 212°F at 98.8 hours, it is conservative to assign the maximum allowed leakage of 0.11 gpm to the faulted SG and 0.19 gpm to the unaffected SGs. The leakage rate of 150 gallons per day converts to a leak rate of approximately 0.104 gpm. Therefore, the assumption of 0.11 gpm through the affected SG is conservative.

8. Regulatory Position 5.2 - The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with leak rate TS.

9. Regulatory Position 5.3 - The primary to secondary leak rate is assumed to continue until primary system pressure is less than secondary system pressure, or until the temperature of the leakage is less than 212°F at 98.8 hours. The release of radioactivity from the unaffected SGs is assumed to continue until shutdown cooling is in operation and releases from the SGs have been terminated.

10. Regulatory Position 5.4 - All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

11. Regulatory Position 5.5.1 – During periods of SG dryout, 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 SGs used for plant cooldown, the primary to secondary leakage was assumed to mix with the secondary water without flashing during periods of total tube submergence.

12. Regulatory Position 5.5.2 - Postulated leakage that immediately flashes to vapor is assumed to rise through the bulk water of the SG into the steam space and is assumed to be immediately released to the environment with no mitigation, i.e., no reduction for scrubbing within the SG bulk water is credited.

13. Regulatory Position 5.5.3 - Leakage that does not immediately flash is assumed to mix with the bulk water.

14. Regulatory Position 5.5.4 - The radioactivity within 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 is assumed. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for particulate radionuclides (alkali metals). Note that for a moisture carryover rate of 0.25%, the partition coefficient would be 400.

15. Regulatory Position 5.6 – SG dryout is not postulated for the intact SGs.

16. RG 1.183 does not address secondary coolant equilibrium specific activity for the alkali metals. This analysis assumes that the equilibrium specific activity of the alkali metals

resulting from SG leakage from the RCS is assumed to be 10% of the primary coolant equilibrium concentration.

17. The integrated mass release values for various time periods assumed in the analysis are provided in Attachment VIII, Table 2, "MSLB Integrated Mass Releases." Mass and associated activity release between these time periods is assumed to be linear. The initial activity release associated with the MSLB is relatively small. The linear flow and activity release rate is conservative because a greater proportion of the mass would be released earlier within the time period, while the activity concentration would be highest at the end of the time period.

18. The ratio of radioiodines to other radionuclides provided in the UFSAR, Table 11.1.1-2, is assumed to be a constant.

19. The analysis assumes that RCS activity conservatively remains constant throughout the pre-accident iodine spike and the fuel failure cases, i.e., no dilution of the RCS activity from the SI system is considered. Additionally, this evaluation assumes that the RCS mass remains constant throughout the MSLB event (no change in the RCS mass is assumed as a result of the MSLB or from the SI system). For the accident-induced iodine spike case, a similar assumption is made with one exception. The primary coolant iodine activity increases during the first eight hours of the transient as a result of the release from the defective fuel at a rate 500 times the iodine equilibrium appearance rate consistent with an initial DE I-131 concentration twice the value of the proposed TS 3.4.16 limits.

20. SG volume in the unaffected SGs is assumed to remain constant throughout the event. Dilution by incoming Auxiliary Feedwater (AFW) is not considered.

21. Releases from the faulted main steam line (and associated SG) are postulated to occur from the nearest main steam line. Releases from the unaffected SGs are postulated from the nearest SG PORV. Based on the atmospheric dispersion values for these two locations provided in Attachment V, Table 3, all releases are conservatively postulated from the closest MSSV/PORV.

22. Data used to calculate the iodine equilibrium appearance rate are provided in Attachment VIII, Table 3, "Iodine Equilibrium Appearance Assumptions."

Methodology:

Input assumptions used in the dose consequence analysis of the MSLB are provided in Attachment VIII, Table 1, "Analysis Inputs/Assumptions." The postulated accident assumes a double-ended break of one main steam line outside containment with the reactor operating at 2346 MWt. The radiological consequences of such an accident bound those of a MSLB accident inside containment. Upon a MSLB, the affected SG rapidly depressurizes and releases the initial contents of the SG to the environment. The rapid secondary depressurization causes a reactor power transient, resulting in a reactor trip. Plant cooldown is achieved via the remaining unaffected SGs.

The analysis assumes that the entire fluid inventory from the affected SG is released to the environment immediately. The affected SG inventory includes the Feedwater (FW) system mass entering the SG prior to FW isolation. The FW mass activity is assumed to be the same as the SG steam activity. AFW mass is not included in the affected SG release inventory because the system takes suction from the condensate storage tank and is assumed to be void of activity. Additional activity, based on the TS 3.4.13 primary to secondary leakage limits, is released via the unaffected SGs via system relief valves until the Residual Heat Removal (RHR) system is placed in operation to continue heat removal from the primary system.

The UFSAR, Section 15.1.5, does not postulate fuel damage for the MSLB accident. Consistent with RG 1.183, Appendix E, Regulatory Position 2, if no or minimal fuel damage is postulated for the limiting event, the activity release is assumed as the maximum allowed by TSs for two cases of iodine spiking: (1) maximum pre-accident iodine spike; and (2) maximum accident-induced, or concurrent, iodine spike.

For the case of a pre-accident iodine spike, a reactor transient is assumed to have occurred prior to the postulated MSLB. The primary coolant iodine concentration is assumed as the maximum value of 60 μCi/gm DE I-131 permitted by the proposed TS 3.4.16. Primary coolant is released into the affected SG by leakage across the SG tubes. The secondary coolant iodine concentration is assumed as the maximum value of 0.1 μCi/gm DE I-131 permitted by TS 3.7.15. Activity is released to the environment from the affected SG, as a result of the postulated primary to secondary leakage and the postulated activity levels of the primary and secondary coolants, until the faulted SG is completely isolated at 98.8 hours (primary system temperature less than 212°F). Primary to secondary leakage is also postulated to occur in the unaffected SGs. Activity is released via steaming from the unaffected SG PORVs until the decay heat generated in the reactor core can be removed by the RHR system at 53.2 hours into the MSLB event. These release assumptions are consistent with the requirements of RG 1.183.

For the case of the accident-induced iodine spike, the postulated MSLB event induces an iodine spike. The activity from the fuel to the reactor coolant is released at a rate of 500 times the iodine equilibrium release rate, which is consistent with twice the Limiting Condition for Operation of the proposed TS 3.4.16 limit of 0.25 μCi/gm DE I-131 for a period of 8 hours. Other release assumptions for this case are identical to those for the pre-accident iodine spike case.

In addition to the non-failed fuel cases described above, a failed fuel case in which two fuel assemblies are breached is considered. A radial peaking factor of 1.80 is assumed. The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant with source term form and release fractions in accordance with Regulatory Position 3.2 of RG 1.183. Other release assumptions for this case are identical to those for the pre-accident iodine spike case except that, prior to the event, the plant is operating at an iodine concentration corresponding to twice the Limiting Condition for Operation of the proposed TS 3.4.16 limit of 0.25 μCi/gm DE I-131.

For this event, the Control Room is automatically placed in the emergency pressurization ventilation mode upon an SI signal. The MSLB accident dose consequence analysis assumes the Control Room emergency pressurization ventilation mode is actuated at 50 seconds to account for emergency diesel generator start, load sequencing, fan start, and damper operation upon a Loss of Offsite Power (LOOP) coincident with the MSLB accident. During normal and emergency pressurization modes the Control Room ventilation intake rate is 400 cfm. During the postulated accident's first hour, an unfiltered inleakage rate of 300 cfm is assumed, including an identified 70 cfm Hagan Room inleakage in the event of a postulated Auxiliary Building exhaust fan failure. Plant procedures preclude Hagan Room inleakage into the Control Room beyond one hour of operation of the emergency pressurization ventilation mode. Therefore, after one hour the assumed Control Room unfiltered inleakage rate is reduced to 230 cfm. Prior to the switchover to the emergency pressurization mode, the activity is assumed to enter the Control Room based on the ventilation intake rate of 400 cfm and Control Room inleakage rate of 300 cfm. Following the switchover to the emergency pressurization mode, the Control Room air is recirculated through filters at a rate of 2600 cfm, and the intake flow is also directed through the filters. Filter removal in the emergency pressurization mode of the Control Room ventilation system is simulated using conservative assumptions based on plant design data as listed in Attachment VII, Table 1.

Radiological Consequences:

The point release-receptor locations are chosen to minimize the distance from the release point to the Control Room intake. Releases from the faulted main steam line (and SG) are postulated from the nearest main steam line to the Control Room. Releases from the unaffected SGs are postulated from the closest safety relief valve to the Control Room. For conservatism, releases are from the release path with the highest values for this accident. Therefore, the closest MSSV/PORV - Control Room release-receptor combination is assumed.

For the Exclusion Area Boundary (EAB) dose calculation, the χ/Q factor for the zero to two hours time interval was assumed for all time periods. Using the zero to two hours χ/Q factor provides a more conservative determination of the EAB dose, because the χ/Q factor for this time period is higher than for any other time period. The Low Population Zone (LPZ) dose was determined using the χ/Q factors provided in Attachment V, Table 1, for the appropriate time intervals.

The radiological consequences of the MSLB Accident are analyzed using the RADTRAD code and the inputs/assumptions previously discussed. Cases corresponding to the reactor coolant maximum pre-accident iodine spike, maximum accident-induced iodine spike, and fuel failures are analyzed. As shown in Attachment VIII, Table 4, "Dose Consequences," the results of the three cases for EAB dose, LPZ dose, and Control Room dose are within the appropriate regulatory acceptance criteria.

Examination of the relatively low dose consequences associated with the MSLB, and comparison to the dose consequences of the LBLOCA event, shows that the LBLOCA dose consequences are more severe than the MSLB event results for EAB, LPZ, and Control Room dose. On this basis, the TSC/EOF was not specifically evaluated for the MSLB using the AST, but is considered to be bounded by the LBLOCA dose consequences.

Reactor Coolant Pump Shaft Seizure (Locked Rotor)

Background:

This event is caused by an instantaneous seizure of a reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Fuel damage may be predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed SG tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere from the secondary coolant system through the SG PORVs. In addition, radioactivity is contained in the primary and secondary coolant before the accident, and some of this activity is released to the atmosphere as a result of steaming from the SGs following the accident. This event is described in the UFSAR, Section 15.3.2.

Compliance with RG 1.183 Regulatory Positions:

The revised Locked Rotor dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix G, "Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident," as discussed below:

1. Regulatory Position 1 - The core inventory of the radionuclide groups utilized for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided as Attachment VI, Table 2. The fraction of fission product inventory in the gap available for release due to fuel breach is consistent with Table 3 of RG 1.183.
2. Regulatory Position 2 – This analysis assumes fuel damage. Therefore, the radiological consequences have been determined.
3. Regulatory Position 3 – Activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.
4. Regulatory Position 4 - The chemical form of the radioiodine released from the damaged fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to equilibrium iodine concentrations in the RCS and secondary system.
5. Regulatory Position 5.1 - The primary to secondary leak rate in the SGs is reduced from the current TS value of 1 gpm through all SGs to a value of 0.3 gpm through all SGs. This submittal proposes to revise the limits of TS 3.4.13 to this new value, as discussed later in this submittal.

6. Regulatory Position 5.2 - The density used in converting volumetric leak rates to mass leak rates is assumed to be 62.4 lbm/ft^3.
7. Regulatory Position 5.3 - The release of radioactivity is assumed to continue until shutdown cooling is in operation and releases from the SGs are terminated.
8. Regulatory Position 5.4 - The analysis assumes a coincident LOOP, which is assumed to occur at the time of the reactor trip. This drives the release from the secondary coolant system through the SG PORVs because condenser cooling is lost.
9. Regulatory Position 5.5 –Noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.
10. Regulatory Position 5.6 – The iodine and transport model for release from the SGs is as follows:
    - Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. Since there is no SG failure assumed in the Locked Rotor analysis, portions of Appendix E, Regulatory Positions 5.5 and 5.6, are not applicable. Specifically, those assumptions associated with SG dryout and tube uncovery are not considered in the analysis.
    - The primary to secondary leakage to the SGs is assumed to mix instantaneously and homogeneously with the secondary water without flashing.
    - The radioactivity in the secondary water is assumed to become a vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed. Although RG 1.183, Appendix E, Footnote 3, defines the partition coefficient in terms of I_2 (elemental iodine), it is assumed that the factor of 100 is an overall partition coefficient and is applicable to all iodine species. It is also assumed that the partition coefficient for the alkali metals is 100, similar to the halogens.
11. RG 1.183, Regulatory Position 3.6 – The amount of fuel damage caused by non-LOCA events is analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. This analysis relies upon departure from nucleate boiling ratio (DNBR) as the fuel damage criterion for estimating fuel damage for the purpose of establishing radioactivity releases. For the Locked Rotor event, Table 3 of RG 1.183 specifies noble gas, alkali metal, and iodine fuel gap release fractions for the breached fuel.
12. RG 1.183 does not address secondary coolant equilibrium specific activity for the alkali metals. This analysis assumes that the equilibrium specific activity of the alkali metals resulting from SG leakage from the RCS is assumed to be 10% of the primary coolant equilibrium concentration.
13. The ratio of radioiodines to other radionuclides provided in the UFSAR, Table 11.1.1-2, is assumed to be a constant.

14. The analysis assumes that the fuel damage is limited to 17 breached fuel assemblies, as a conservative assumption. The current licensing basis assumes that fuel damage is limited to eight fuel assemblies.
15. AFW flow to the SGs is assumed to occur at T=0. This source of makeup water is assumed to be void of radioactivity. An evaluation showed that flow initiated at T=10 minutes resulted in no difference in offsite doses.

Methodology:

Input assumptions used in the dose consequence analysis of the Locked Rotor event are provided in Attachment IX, Table 1, "Analysis Inputs/Assumptions." This event is caused by an instantaneous seizure of a reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Following reactor trip, heat stored in the fuel rods continues to be transferred to the reactor coolant. Because of the reduced core flow, coolant temperatures will rise. The rapid rise in primary system temperatures during the initial phase of the transient results in a reduction in the initial Departure from Nucleate Boiling (DNB) margin and fuel damage.

Based on limiting assumptions, a total of 17 fuel assemblies are assumed to fail by clad breach. A radial peaking factor of 1.80 is assumed. The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant with source term form and release fractions consistent with Appendix G of RG 1.183. Primary coolant is released into the SGs as a result of postulated primary to secondary leakage. Activity is released to the atmosphere via steaming from the SG PORVs until the decay heat generated in the reactor core can be removed by the RHR system at 53.2 hours into the event. These release assumptions are consistent with the requirements of RG 1.183.

For this event, the Control Room is automatically placed in the emergency pressurization ventilation mode upon a Control Room area radiation monitor isolation signal. For analysis purposes, the switchover to the emergency pressurization mode is assumed to occur at one hour. During the postulated accident's first hour, an unfiltered inleakage rate of 300 cfm is assumed, including an identified 70 cfm Hagan Room inleakage in the event of a postulated Auxiliary Building exhaust fan failure. Plant procedures preclude Hagan Room inleakage into the Control Room beyond one hour of operation of the emergency pressurization ventilation mode. Therefore, after one hour the assumed Control Room unfiltered inleakage rate is reduced to 230 cfm. Prior to the switchover to the emergency pressurization mode, the activity is assumed to enter the Control Room based on the normal ventilation intake rate of 400 cfm and the Control Room inleakage rate of 300 cfm. Following the switchover to the emergency pressurization mode, the Control Room air is recirculated through filters at a rate of 2600 cfm, and the intake flow is also directed through the filters. Filter removal in the emergency pressurization mode of the Control Room ventilation system is simulated using conservative assumptions based on plant design data as listed in Attachment VII, Table 1.

Radiological Consequences:

The point release-receptor locations are chosen to minimize the distance from the release point to the Control Room intake. Therefore, the closest MSSV/PORV - Control Room release-receptor combination is assumed.

For the EAB dose calculation, the χ/Q factor for the zero to two hours time interval was assumed for all time periods. Using the zero to two hours χ/Q factor provides a more conservative determination of the EAB dose, because the χ/Q factor for this time period is higher than for any other time period. The LPZ dose was determined using the χ/Q factors provided in Attachment V, Table 1, for the appropriate time intervals.

The radiological consequences of the Locked Rotor event are analyzed using the RADTRAD code and the inputs/assumptions previously discussed. As shown in Attachment IX, Table 3, "Dose Consequences," the radiological consequences of the Locked Rotor event for the EAB, LPZ, and Control Room are within the appropriate regulatory acceptance criteria.

Examination of the relatively low dose consequences associated with the Locked Rotor event, and comparison to the dose consequences of the LBLOCA event, shows that the LBLOCA dose consequences are more severe than the Locked Rotor event results for EAB, LPZ, and Control Room doses. On this basis, the TSC/EOF was not specifically evaluated for the Locked Rotor using the AST, but is considered to be bounded by the LBLOCA dose consequences.

Single Rod Control Cluster Assembly (RCCA) Withdrawal

Background:

This event is initiated by the inadvertent withdrawal of a single RCCA at power. The resulting insertion of positive reactivity results in a power excursion transient. Fuel damage may be predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed SG tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere from the secondary coolant system through the SG PORVs. In addition, radioactivity is contained in the primary and secondary coolant before the accident, and some of this activity is released to the atmosphere as a result of steaming from the SGs following the accident. This event is described in the UFSAR, Section 15.4.3.1.

Compliance with RG 1.183 Regulatory Positions:

Although RG 1.183 does not specifically address the Single RCCA Withdrawal event, the RG is utilized as a guide for the analysis of dose consequences from this postulated event. The Locked Rotor event release path, described in RG 1.183, Appendix G, most closely resembles that of the Single RCCA Withdrawal event. Fuel damage is treated in a manner similar to the LBLOCA, because, unlike the Locked Rotor event, the Single RCCA Withdrawal event considers fuel damage. Therefore, this dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix G, and RG 1.183, Section 3.2, Table 2. Regulatory Positions associated with the Single RCCA Withdrawal event are discussed below:

1. Regulatory Position 1 – The core inventory of the radionuclide groups utilized for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided as Attachment VI, Table 2. The release from the damaged fuel is based on the estimate of the number of fuel rods breached, the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting, and the radial peaking factor. Release fractions are consistent with RG 1.183, Section 3.2, Table 3 and Section 3.2, Table 2, as appropriate.
2. Regulatory Position 2 – This analysis assumes fuel damage. Therefore, the radiological consequences have been determined.
3. Regulatory Position 3 – Activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.
4. Regulatory Position 4 - The chemical form of the radioiodine released from the damaged fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to equilibrium iodine concentrations in the RCS and secondary system.
5. Regulatory Position 5.1 - The primary to secondary leak rate in the SGs is reduced from the current TS value of 1 gpm through all SGs to a value of 0.3 gpm through all SGs. This submittal proposes to revise TS 3.4.13 to this new value, as discussed later in this submittal.
6. Regulatory Position 5.2 - The density used in converting volumetric leak rates to mass leak rates is assumed to be 62.4 lbm/ft^3.
7. Regulatory Position 5.3 - The release of radioactivity is assumed to continue until shutdown cooling is in operation and releases from the SGs are terminated.
8. Regulatory Position 5.4 - The analysis assumes a coincident LOOP, which is assumed to occur at the time of the reactor trip. This drives the release from the secondary coolant system through the SG PORVs or safety valves, because condenser cooling is lost.
9. Regulatory Position 5.5 –Noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.
10. Regulatory Position 5.6 – The iodine and transport model for release from the SGs is as follows:
    - Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. Since there is no SG failure assumed in the Single RCCA Withdrawal analysis,

portions of Appendix E, Regulatory Positions 5.5 and 5.6, are not applicable. Specifically, those assumptions associated with SG dryout and tube uncovery are not considered in the analysis.

- The primary to secondary leakage to the SGs is assumed to mix instantaneously and homogeneously with the secondary water without flashing.
- The radioactivity in the secondary water is assumed to become a vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed. Although RG 1.183, Appendix E, Footnote 3, defines the partition coefficient in terms of I_2 (elemental iodine), it is assumed that the factor of 100 is an overall partition coefficient and is applicable to all iodine species. It is also assumed that the partition coefficient for the alkali metals is 100, similar to the halogens.

11. RG 1.183 does not address secondary coolant equilibrium specific activity for the alkali metals. This analysis assumes that the equilibrium specific activity of the alkali metals resulting from SG leakage from the RCS is assumed to be 10% of the primary coolant equilibrium concentration.
12. The ratio of radioiodines to other radionuclides provided in the UFSAR, Table 11.1.1-2, is assumed to be a constant.
13. The analysis assumes that the fuel damage is limited to one breached fuel assembly and three assemblies that reach or exceed the initiation temperature of fuel melt, as a conservative assumption. The current licensing basis assumes that fuel damage is limited to three fuel assemblies.
14. AFW flow to the SGs is assumed to occur at T=0. This source of makeup water is assumed to be void of radioactivity. An evaluation showed that flow initiated at T=10 minutes resulted in no difference in offsite doses.

Methodology:

Input assumptions used in the dose consequence analysis of the Single RCCA Withdrawal event are provided in Attachment X, Table 1, "Analysis Inputs/Assumptions." The event is initiated by the inadvertent withdrawal of a single control rod at power. The ensuing reactivity insertion causes core power to increase. In the event that the secondary steam dump control system does not respond to the increased power production, secondary system temperature and pressure will increase, causing a corresponding increase in primary coolant temperature. This increase in primary coolant temperature occurs slowly enough that the pressurizer pressure control system, if available, is capable of suppressing the primary pressure increase. The degradation of coolant conditions coupled with the power increase may result in approaching DNB conditions in the hot channel. High radial power peaking is quite localized in the region of the single withdrawn RCCA and may, in severe cases, surpass the design limits. Thus, assemblies in the immediate vicinity of the withdrawn RCCA may experience boiling transition. Although such exposure would be limited to short time periods, some fuel damage might occur.

Based on limiting assumptions, a total of three fuel assemblies are assumed to reach or exceed the initiation temperature of fuel melt and one additional assembly is assumed to be breached. A radial peaking factor of 1.80 is assumed.

The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant with source term form and release fractions consistent with Appendix G of RG 1.183. Primary coolant is released into the SGs as a result of primary to secondary leakage. Activity is released to the atmosphere via steaming from the SG PORVs until the decay heat generated in the reactor core can be removed by the RHR system at 53.2 hours into the event. These release assumptions are consistent with the requirements of RG 1.183.

For this event, the Control Room is automatically placed in the emergency pressurization ventilation mode upon a Control Room area radiation monitor isolation signal. For analysis purposes, the switchover to the emergency pressurization mode is assumed to occur at one hour. During the postulated accident's first hour, an unfiltered inleakage rate of 300 cfm is assumed, including an identified 70 cfm Hagan Room inleakage in the event of a postulated Auxiliary Building exhaust fan failure. Plant procedures preclude Hagan Room inleakage into the Control Room beyond one hour of operation in the emergency pressurization ventilation mode. Therefore, after one hour the assumed Control Room unfiltered inleakage rate is reduced to 230 cfm. Prior to the switchover to the emergency pressurization mode, the activity is assumed to enter the Control Room based on the normal ventilation intake rate of 400 cfm and the Control Room inleakage rate of 300 cfm. Following the switchover to the emergency pressurization mode, the Control Room air is recirculated through filters at a rate of 2600 cfm, and the intake flow is also directed through the filters. Filter removal in the emergency pressurization mode of the Control Room ventilation system is simulated using conservative assumptions based on plant design data as listed in Attachment VII, Table 1.

Radiological Consequences:

The point release-receptor locations are chosen to minimize the distance from the release point to the Control Room intake. Releases from the SGs are postulated from the nearest safety relief valve to the Control Room. Therefore, the closest MSSV/PORV – Control Room release-receptor combination is assumed.

The radiological consequences of the Single RCCA Withdrawal event are analyzed using the RADTRAD code and the inputs/assumptions previously discussed. The regulatory requirement in RG 1.183, Regulatory Position 4.1.5, is that the "worst 2 hour interval" dose be limited to 2.5 rem TEDE. Although the X/Q factors vary over the duration of the accident (see Attachment V, Tables 1 and 4), the EAB dose is based on the worst two hour time interval. Based on the RADTRAD outputs, the peak two hour EAB dose begins at various times for the five sources, as shown in Attachment X, Table 3. RG 1.183, Regulatory Position 4.15 allows doses that occur "outside" the time interval of the major contributing source to be reduced by the ratio of the X/Q factor at zero to two hours divided by the X/Q factor corresponding to the time interval between the source with the greatest EAB dose and the source in question.

The halogen RCS source results in the greatest EAB dose. Conservatively assuming that the EAB doses resulting from the halogen and alkali metal RCS sources occur at approximately the same time, then the EAB doses for the noble gas RCS source and the halogen and alkali metal secondary coolant system source can be adjusted to reflect a X/Q factor (4.11E-04 sec/m^3) for the one to four day time period (since the time interval is >24 hours). Using the correction method discussed above, the EAB dose for the worst two hour interval is corrected to remove excess conservatism. As shown in Attachment X, Table 4, "Dose Consequences," the radiological consequence of the Single RCCA Withdrawal event for the EAB, LPZ, and Control Room are within the appropriate regulatory acceptance criteria.

Examination of the relatively low dose consequences associated with the Single RCCA Withdrawal event, and comparison to the dose consequences of the LBLOCA event, shows that the LBLOCA dose consequences are more severe than the Single RCCA Withdrawal event results for EAB, LPZ, and Control Room doses. On this basis, the TSC/EOF was not specifically evaluated for the Single RCCA Withdrawal event using the AST, but is considered to be bounded by the LBLOCA dose consequences.

Steam Generator Tube Rupture (SGTR)

Background:

This event is assumed to be caused by the instantaneous rupture of a SG tube, which relieves to the lower pressure secondary. No fuel damage is postulated for the HBRSEP, Unit No. 2, SGTR event. The SGTR event is described in the UFSAR, Section 15.6.3.

Compliance with RG 1.183 Regulatory Positions:

The revised SGTR dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix F, "Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident," as discussed below:

1. Regulatory Position 1 – The core inventory of the radionuclide groups utilized for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided as Attachment VI, Table 2.
2. Regulatory Position 2 – No fuel damage is postulated to occur for the HBRSEP, Unit No. 2, SGTR event. Two cases of iodine spiking have been assumed.
3. Regulatory Position 2.1 – One case of iodine spiking assumes a reactor transient prior to the postulated SGTR that raises the primary coolant iodine concentration to the maximum allowed by the proposed TS 3.4.16 value of 60.0 μCi/gm DE I-131. This is the pre-accident spike case.

4. Regulatory Position 2.2 – One case of iodine spiking assumes the transient associated with the SGTR causes an iodine spike. The spiking model assumes the primary coolant activity is initially at the proposed TS 3.4.16 value of 0.25 µCi/gm DE I-131. Iodine is assumed to be released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of eight hours. This is the accident-induced case.
5. Regulatory Position 3 – The activity released from the fuel is assumed to be instantaneously and homogeneously mixed through the primary system.
6. Regulatory Position 4 – Iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic.
7. Regulatory Position 5.1 – The primary to secondary leak rate in the SGs is assumed to be the leak rate specified in the proposed TS 3.4.13. The primary to secondary leak rate is apportioned between the SGs in such a manner that the calculated dose is maximized. The proposed TS 3.4.13 requirements limit leakage through all SGs to 0.3 gpm and leakage through any one SG to 150 gallons per day. Since the majority of steam release during cooldown will occur in the two intact SGs, it is conservative to assign the maximum allowed leakage of 0.11 gpm to each of the unaffected SGs and the remainder to the ruptured SG. The leakage rate of 150 gallons per day converts to a leak rate of approximately 0.104 gpm. Therefore, the assumption of 0.11 gpm through each of the unaffected SGs is conservative.
8. Regulatory Position 5.2 – The density used in converting volumetric leak rates to mass leak rates is assumed to be 62.4 lbm/ft^3.
9. Regulatory Position 5.3 – The primary to secondary leakage is 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 212°F. The release of radioactivity is assumed to continue until the RHR system is placed in operation and releases from the SGs are terminated.

   The current licensing basis for termination of the affected SG activity release is based on the original analyses described in the original Final Safety Analysis Report, Section 14.2.4. This basis was maintained, with adjustments for the replacement SGs, by scaling analyses performed by Exxon Nuclear (now Framatome-ANP), as currently described in the UFSAR, Section 15.6.3. For the purposes of the AST dose consequence analysis, the HBRSEP, Unit No. 2, current licensing basis, which states that the affected SG will be isolated within 30 minutes by operator action, will be maintained. This isolation terminates releases from the affected SG, while primary to secondary leakage continues to provide activity for release from the two unaffected SGs.
10. Regulatory Position 5.4 – The release of fission products from the secondary system is evaluated with the assumption of a coincident LOOP.
11. Regulatory Position 5.5 –Noble gases released from the primary system are assumed to be released to the environment without reduction or mitigation.

12. Regulatory Position 5.6 – Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine and transport model for release from the SGs is as follows:

    - SG dryout is not postulated in the HBRSEP, Unit No. 2, SGTR event.
    - A portion of the primary to secondary leakage through the SGTR is assumed to flash to vapor, based on the thermodynamic conditions in the RCS and secondary. The leakage that immediately flashes to vapor is assumed to rise through the bulk water of the SG, enters the steam space, and is assumed to be immediately released to the environment with no mitigation, i.e., no reduction for scrubbing within the SG bulk water is credited.
    - Leakage that does not immediately flash is assumed to mix with the bulk water.
    - The radioactivity in the secondary water is assumed to become a vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed.

13. RG 1.183 does not address secondary coolant equilibrium specific activity for the alkali metals. This analysis assumes that the equilibrium specific activity of the alkali metals resulting from SG leakage from the RCS is assumed to be 10% of the primary coolant equilibrium concentration.

14. The ratio of radioiodines to other radionuclides provided in the UFSAR, Table 11.1.1-2, is assumed to be a constant.

15. The analysis assumes that the RCS activity conservatively remains constant throughout the pre-accident iodine spike case, i.e., no dilution of the RCS activity from the SI system is considered. Additionally, the evaluation assumes that RCS mass remains constant throughout the SGTR event (no change in RCS mass is assumed as a result of the rupture flow within the SGTR or from the SI system).

    For the accident-induced iodine spike case, a similar assumption is made with one exception. The primary coolant iodine activity increases during the first eight hours of the transient as a result of the release from the defective fuel at a rate 335 times the iodine equilibrium appearance rate consistent with an initial DE I-131 concentration twice the value of the proposed TS 3.4.16 limits.

16. SG volume is assumed to remain constant throughout both the pre-accident and the accident-induced iodine spike SGTR events. Dilution by incoming AFW is not considered.

17. Data used to calculate the iodine equilibrium appearance rate are provided in Attachment XI, Table 3, "Iodine Equilibrium Appearance Assumptions."

Methodology:

Input assumptions used in the dose consequence analysis of the SGTR event are provided in Attachment XI, Table 1, "Analysis Inputs/Assumptions." This event is assumed to be caused by the instantaneous rupture of a SG tube releasing primary coolant to the lower pressure secondary system. In the unlikely event of a concurrent loss of power, the loss of circulating water through the condenser would eventually result in the loss of condenser vacuum. The main steam bypass valves would automatically close to protect the condenser, thereby causing steam relief directly to the atmosphere from the SG PORVs or MSSVs. This direct steam relief would continue until the ruptured SG is isolated. The isolation is assumed to require 30 minutes. The SG is isolated on the secondary side by closing associated inlet and outlet secondary valves.

A thermal-hydraulic analysis was performed to determine a conservative maximum break flow, break flashing fraction, and steam release inventory through the ruptured SG relief valves. Additional activity, based on the proposed TS 3.4.13 primary to secondary leakage limits, is released via the unaffected SGs via the secondary PORVs until the RHR system is placed in operation to continue heat removal from the primary system.

The HBRSEP, Unit No. 2, UFSAR, Section 15.6.3, postulates no fuel damage for the SGTR. Consistent with RG 1.183, Appendix F, Regulatory Position 2, if no or minimal fuel damage is postulated for the limiting event, the activity release is assumed as the maximum allowed by TSs for two cases of iodine spiking: (1) maximum pre-accident iodine spike and (2) maximum accident-induced, or concurrent, iodine spike.

For the case of a pre-accident iodine spike, a reactor transient is assumed to have occurred prior to the postulated SGTR event. The primary coolant iodine concentration has been raised to the maximum value of 60 μCi/gm DE I-131 permitted by the proposed TS 3.4.16. Primary coolant is released into the ruptured SG by the tube rupture and by a fraction of the total proposed TS 3.4.13 allowable primary to secondary leakage. Activity is released to the environment from the ruptured SG via direct flashing of a fraction of the released primary coolant from the tube rupture and also via steaming from the ruptured SG PORV until the ruptured SG is isolated at 30 minutes. The unaffected SGs are used to cool down the plant during the SGTR event. Primary to secondary tube leakage is also postulated into the unaffected SGs. Activity is released via steaming from the unaffected SG PORVs until the decay heat generated in the reactor core can be removed by the RHR system at 53.2 hours into the SGTR event. These release assumptions are consistent with the requirements of RG 1.183.

For the case of the accident-induced iodine spike, the postulated SGTR event induces an iodine spike. RCS activity is initially assumed to be 0.25 μCi/gm DE I-131, consistent with the proposed TS 3.4.16. Iodine is released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. Other release assumptions for this case are identical to those for the pre-accident iodine spike case.

For this event, the Control Room is automatically placed in the emergency pressurization ventilation mode upon a Control Room area radiation monitor isolation signal. The SGTR analysis assumes the realignment to the emergency pressurization ventilation mode occurs at 310 seconds to account for fan start, damper operation, and dose rates inside the Control Room above the area radiation monitor setpoint. Prior to this switchover, the activity is assumed to enter the Control Room at the normal intake rate of 400 cfm. During the postulated accident's first hour, an unfiltered inleakage rate of 300 cfm is assumed, including an identified 70 cfm Hagan Room inleakage in the event of a postulated Auxiliary Building exhaust fan failure. Plant procedures preclude Hagan Room inleakage into the Control Room beyond one hour of operation in the emergency pressurization ventilation mode. Therefore, after one hour the assumed Control Room unfiltered inleakage rate is reduced to 230 cfm. Prior to the switchover to the emergency pressurization mode, the activity is assumed to enter the Control Room based on the normal ventilation intake rate of 400 cfm and the Control Room inleakage rate of 300 cfm. Following the switchover to the emergency pressurization mode, the Control Room air is recirculated through filters at a rate of 2600 cfm, and the intake flow is also directed through the filters. Filter removal in the emergency pressurization mode of the Control Room ventilation system is simulated using conservative assumptions based on plant design data as listed in Attachment VII, Table 1.

Radiological Consequences:

The point release-receptor locations are chosen to minimize the distance from the release point to the Control Room intake. Releases from the ruptured and unaffected SGs are postulated from the nearest safety relief valve to the Control Room. Therefore, the closest MSSV/PORV – Control Room release-receptor combination is assumed.

For the EAB dose calculation, the χ/Q factor for the zero to two hours time interval was assumed for all time periods. Using the zero to two hours χ/Q factor provides a more conservative determination of the EAB dose, because the χ/Q factor for this time period is higher than for any other time period. The LPZ dose was determined using the χ/Q factors provided in Attachment V, Table 1, for the appropriate time intervals.

The radiological consequences of the SGTR accident are analyzed using the RADTRAD code and the inputs/assumptions previously discussed. Two activity release cases corresponding to the reactor coolant maximum pre-accident iodine spike and maximum accident-induced iodine spike, based on the proposed TS 3.4.16 limits, are analyzed. As shown in Attachment XI, Table 4, "Dose Consequences," the radiological consequences of the SGTR event for the EAB, LPZ, and Control Room are within the appropriate regulatory acceptance criteria.

Examination of these dose consequence results, and comparison to the dose consequence results of the LBLOCA event, shows that the LBLOCA dose consequences and the SGTR dose consequences are comparable for EAB, LPZ, and Control Room doses. On this basis, the TSC/EOF was not specifically evaluated for the SGTR using the AST, but is considered to be bounded by the LBLOCA dose consequences.

Large Break Loss-of-Coolant Accident (LBLOCA)

Background:

This event is assumed to be caused by an abrupt failure of the main reactor coolant pipe and the Emergency Core Cooling System (ECCS) fails to prevent the core from experiencing significant degradation. This sequence cannot occur unless there are multiple failures, and thus is beyond the typical design basis accident that considers a single active failure. Activity from the core is released to the containment and from there is released to the environment by means of containment leakage and leakage from the ECCS. The Loss-of-Coolant Accidents are discussed in UFSAR, Section 15.6.5.

Compliance with RG 1.183 Regulatory Positions:

The revised LBLOCA dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix A, "Assumptions for Evaluating the Radiological Consequences of a LWR Loss-Of-Coolant Accident," as discussed below:

1. Regulatory Position 1 - The core inventory of the radionuclide groups utilized for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided as Attachment VI, Table 1. The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel phases of the LBLOCA, are based on RG 1.183, Regulatory Position 3.2, Table 2, and are provided as Attachment VI, Table 5.
2. Regulatory Position 2 - The sump pH is controlled at a value greater than 7.0. HBRSEP, Unit No. 2, plant procedures provide guidance to maintain the sump pH above 8.5 through the use of Sodium Hydroxide (NaOH) from the Spray Additive Tank. Therefore, the chemical form of radioiodine released to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. With the exception of elemental iodine, organic iodide, and noble gases, fission products are assumed to be in particulate form.
3. Regulatory Position 3.1 - The radioactivity released from the fuel is assumed to be mixed instantaneously and homogeneously throughout the free air volume of the containment. The release into the containment is assumed to terminate at the end of the early in-vessel phase.
4. Regulatory Position 3.2 - Reduction in airborne radioactivity by natural deposition within the containment is credited. An acceptable model or user-defined, time-dependent removal coefficients for removal of iodine and aerosols are incorporated into the RADTRAD computer code. A natural deposition removal coefficient of 0.1/hr is assumed. This analysis also conservatively assumes that the iodine aerosol removal by natural deposition is limited to a reduction factor of 1000. It should be noted that natural deposition is credited in both the sprayed and unsprayed regions of the containment. In the sprayed region, natural deposition is only credited when containment spray is not in operation.

5. Regulatory Position 3.3 - Containment spray train 'A' provides coverage to ~52% of the containment. Containment spray train 'B' provides coverage to ~59% of the containment. Conservatively, the RADTRAD model utilized a spray coverage of 52% (less removal of radioactive aerosols and elemental iodine from the containment as a function of time). Consequently, the unsprayed area is ~48% of the containment. Therefore, the HBRSEP, Unit No. 2, containment building atmosphere is not considered to be a single, well-mixed volume.

   The maximum decontamination factor (DF) for the elemental iodine spray removal coefficient is 200 based on the maximum airborne elemental iodine concentration in the containment. Based on the timing of the release phases provided in RG 1.183, Regulatory Position 3.3, Table 4, the expected maximum concentration should occur at 1.8 hours. The analysis confirmed this assumption. The analysis also showed that an elemental iodine DF of 200 is achieved at 2.46 hours.

   The particulate iodine removal rate is reduced by a factor of 10 when a DF of 50 is reached. For conservative reasons, the depletion of particulate iodine by natural deposition is also included in achieving the DF of 50. The analysis showed that a particulate iodine DF of 50 is achieved at 20.8 hours.

6. Regulatory Position 3.4 - Reduction in airborne radioactivity in the containment by filter recirculation systems is not assumed in this analysis.

7. Regulatory Position 3.5 - HBRSEP, Unit No. 2, is not a Boiling Water Reactor (BWR). Therefore, the positions related to suppression pool scrubbing under Regulatory Position 3.5 are not applicable.

8. Regulatory Position 3.6 – The analysis does not assume a reduction in airborne radioactivity by Engineering Safety Features (ESF) not discussed above. The HBRSEP, Unit No. 2, containment is not equipped with an ice condenser.

9. Regulatory Position 3.7 - A containment leak rate of 0.1% by weight of the containment air is assumed for the first 24 hours at Pa of 40.5 psig. After 24 hours, the containment leak rate is reduced to 0.05% by weight of the containment air.

10. Regulatory Position 3.8 – Routine containment purge is not considered in this calculation. Purging during power operations is considered to be an infrequent operation at HBRSEP, Unit No. 2.

11. Regulatory Position 4 - The HBRSEP, Unit No. 2, containment is not a dual containment system. Therefore, the positions related to dual containments under Regulatory Position 4 are not applicable.

12. Regulatory Position 5.1 – ESF systems that recirculate water outside the primary containment are assumed to leak during their intended operation. With the exception of noble gases, all fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the containment sump water at the time of release from the core.

13. Regulatory Position 5.2 – Leakage from the ESF system is taken as two times the value of the HBRSEP, Unit No. 2, Technical Requirements Manual (TRM),

Specification 3.23, "Post Accident Recirculation Heat Removal System Leakage," limit of 2 gph. Leakage sources in this limit include valve stems, flanges, and pump seals. The leakage is 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.

14. Regulatory Position 5.3 – With the exception of the non-particulate iodines, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.
15. Regulatory Positions 5.4 and 5.5 – The flash fraction assumed in the analysis is the current license basis value of 5.3%. As discussed in CP&L letter dated May 21, 1990, this flash fraction is based on a conservative set of assumptions. The sump water is conservatively assumed to remain at 263°F. The NRC found the description of the 30-day post-LOCA Control Room personnel dose analysis to be acceptable in a Safety Evaluation dated October 26, 1990.
16. Regulatory Position 5.6 – The radioiodine that is postulated to be available for release from the sump to the environment is assumed to be 97% elemental and 3% organic. No reduction for dilution or holdup within buildings, or by ESF ventilation filtration systems, is credited.
17. Regulatory Position 6 –HBRSEP, Unit No. 2, is not a BWR. Therefore, the positions related to main steam isolation valve leakage under Regulatory Position 6 are not applicable.
18. Regulatory Position 7 – Containment purge is not considered as a means of combustible gas or pressure control in this analysis.
19. Spray removal coefficients of 20/hr and 3.057/hr are assumed for elemental iodine and particulate iodine, respectively.
20. Two safety-related containment cooling fans are assumed to be operational post-accident, with a flow rate of 65,000 cfm per fan.
21. Due to the location of the Refueling Water Storage Tank (RWST) with respect to the Control Room and TSC/EOF, for conservative reasons, it is assumed that ESF system leakage is from the RHR heat exchanger room (located in the Auxiliary Building), which has a greater χ/Q factor.

Methodology:

The key inputs used in the dose consequence analysis are included in Attachment XII, Table 1, "Analysis Inputs/Assumptions." These inputs and assumptions fall into three main categories, Radionuclide Release Inputs, Radionuclide Transport Inputs, and Radionuclide Removal Inputs.

For the purpose of LBLOCA analyses, a major LOCA is defined as a rupture sized 1.0 ft^2 or larger of the RCS piping, including the double-ended rupture of the largest pipe in the RCS or of any line connected to that system up to the first closed valve. Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer.

A reactor trip signal occurs when the pressurizer low pressure trip setpoint is reached. An SI signal is actuated when the appropriate setpoint (high containment pressure) is reached. The following countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat, and

2. Injection of borated water provides heat transfer from the core and prevents excessive cladding temperatures.

LBLOCA Release Inputs:

The core inventory of the radionuclide groups utilized for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided as Attachment VI, Table 1. The source term represents end-of-cycle conditions assuming a bounding initial fuel enrichment and a core burnup of 60,000 MWD/MTU.

Per TS Surveillance Requirement 3.6.1.1, the leakage rate acceptance criteria for the containment is $\leq$ 1.0 La. TS 5.5.6 defines La as 0.1% of the containment air weight per day at a pressure of 40.5 psig. Therefore, for the first 24 hours, the containment is assumed to leak at a rate of 0.1% of the containment air weight per day. In accordance with RG 1.183, Regulatory Position 3.7, the containment leakage rate is reduced by 50% at 24 hours into the LOCA to 0.05% (by weight) per day.

ESF systems are assumed to leak at a rate of 4 gph into the Auxiliary Building based on TRM Specification 3.23, which limits ESF leakage to 2 gph. This includes a safety factor of 2 as required per RG 1.183. The HBRSEP, Unit No. 2, ESF leakage licensing basis flashing fraction of 5.3% is assumed, and thus 5.3% of the total ESF source iodine activity is assumed airborne. The ESF leakage rate is assumed to start at 21 minutes after the accident and continue throughout the 30-day duration of the postulated accident. This start time is conservative in that it is associated with two ESF train operation, whereas one train, as assumed in the analysis for containment spray operation, would result in a longer system start time due to slower depletion of the available water inventory.

LBLOCA Transport Inputs:

During the LBLOCA event, both the containment and Auxiliary Building ESF leakage are assumed released directly to the environment unfiltered and at ground level. To maximize the calculated post-accident doses, the ground level containment releases are assumed to discharge from the building's closest location to the receptor location at the Control Room or TSC/EOF air intake. The ground level Auxiliary Building ESF leakage is assumed to discharge from the closest ESF equipment location (the RHR heat exchanger room) to the receptor location at the Control Room or the TSC/EOF air intake without credit for Auxiliary Building dilution or holdup.

For this event, the Control Room is automatically isolated and placed in the emergency pressurization ventilation mode upon an SI signal. The LBLOCA analysis assumes the Control Room emergency pressurization ventilation mode is actuated at 35 seconds to account for emergency diesel generator start, load sequencing, fan start, and damper operation upon a LOOP coincident with the LBLOCA. During normal and emergency pressurization modes, the Control Room ventilation intake rate is 400 cfm. During the postulated accident's first hour, an inleakage rate of 170 cfm is assumed, including an identified 70 cfm Hagan Room inleakage in the event of a postulated Auxiliary Building exhaust fan failure. Plant procedures preclude Hagan Room inleakage into the Control Room beyond one hour of operation in the emergency pressurization ventilation mode. Therefore, after one hour the assumed Control Room unfiltered inleakage rate is reduced to 100 cfm. Prior to the switchover to the emergency pressurization mode, the activity is assumed to enter the Control Room based on the normal ventilation intake rate of 400 cfm and the Control Room inleakage rate of 170 cfm. Following the switchover to the emergency pressurization mode, the Control Room air is recirculated through filters at a rate of 2600 cfm, and the intake flow is also directed through the filters. Filter removal in the emergency pressurization mode of the Control Room ventilation system is simulated using conservative assumptions based on plant design data as listed in Attachment VII, Table 1.

The TSC/EOF is automatically placed in the Emergency Air Filtration mode upon an air intake radiation monitor signal. For analysis purposes, the accident activity is allowed to enter the TSC/EOF for the first two hours of the LBLOCA unfiltered and at the normal ventilation flow rate of 3420 cfm, at which time the TSC/EOF is switched to the Emergency Air Filtration mode. Following the switchover to the Emergency Air Filtration mode, the air intake flow rate is maintained at 3420 cfm through the filtration system. An unfiltered TSC/EOF inleakage rate of 500 cfm is assumed in the analysis. Attachment VII, Table 2, "TSC/EOF Ventilation System Parameters," provides those key TSC/EOF ventilation parameters that are assumed in the LBLOCA dose consequence analysis.

LBLOCA Removal Inputs:

The accident activity released from the core is partially removed by natural deposition and spray mechanisms in containment, as well as by air filtration systems in the Control Room and TSC/EOF. Containment natural deposition is credited at all times in the containment unsprayed regions and during periods in which the Containment Spray System (CSS) is not operational in the sprayed region. Safety-related containment air recirculation fans provide mixing between the sprayed and unsprayed regions. The natural deposition removal coefficient is assumed as 0.1/hr. As an additional conservatism, no natural deposition is assumed when a natural deposition DF of 1000 is reached. CSS particulate removal is determined using the model described in Standard Review Plan (SRP) 6.5.2. Only one train of containment spray is credited, and the system start and termination timing is conservatively assumed to bound all modes of operation.

Filter removal in the Control Room emergency pressurization mode and the TSC/EOF Emergency Air Filtration mode is simulated using conservative assumptions based on plant design data as listed in Attachment VII, Tables 1 and 2. An aerosol removal efficiency of 95% and 99% is used for the Control Room and TSC/EOF ventilation system HEPA filters, respectively. Charcoal filter removal efficiencies for the Control Room and TSC/EOF charcoal filters are 95% and 99%, respectively, for both elemental and organic iodine.

Radiological Consequences:

The radiological consequences of the design basis LOCA are analyzed using the RADTRAD code and the inputs/assumptions previously discussed. In addition, the MicroShield code, Version 5.05, Grove Engineering, is used to develop direct shine doses to the Control Room and TSC/EOF. MicroShield is a point kernel integration code used for general purpose gamma shielding analysis. The MicroShield code is not considered to be an NRC-approved code. However, it has been used to support licensing submittals that have been accepted by the NRC (see Duane Arnold Energy Center letter dated October 19, 2000, and the associated NRC Safety Evaluation dated July 31, 2001).

The post accident doses are the result of two distinct activity releases:

1. Containment leakage is directly released into the environment throughout the 30-day accident.

2. ESF system leakage into the Auxiliary Building is directly released into the environment at the location of ESF equipment/piping in the Auxiliary Building.

For the EAB dose calculation, the χ/Q factor for the zero to two hours time interval was assumed for all time periods. Using the zero to two hours χ/Q factor provides a more conservative determination of the EAB dose, because the χ/Q factor for this time period is higher than for any other time period. The LPZ dose was determined using the χ/Q factors provided in Attachment V, Table 1, for the appropriate time intervals.

The dose to both the Control Room and TSC/EOF occupants includes terms for:

1. Inleakage internal cloud immersion and inhalation contribution from the containment and ESF leakage releases.

2. External cloud contribution from the containment and ESF leakage releases. This term takes credit for Control Room and TSC/EOF structural shielding.

3. A direct dose contribution from the containment's contained accident activity. This term takes credit for both containment and Control Room and TSC/EOF structural shielding.

As shown in Attachment XII, Table 3, "Dose Consequences," the radiological consequences of the LBLOCA event for the EAB, LPZ, Control Room, and TSC/EOF are within the appropriate regulatory acceptance criteria.

Waste Gas Decay Tank (WGDT) Rupture

Background:

This event considers the dose consequences of a rupture of a WGDT. Radioactive Waste Gas System leaks and failures are discussed in the UFSAR, Section 15.7.1. The WGDT event is not affected by changes in the source term or by changes in the licensed power level. However, the analysis has been revised to incorporate new χ/Q factors and to evaluate the dose consequences in terms of TEDE.

Compliance with RG 1.183 Regulatory Positions:

Although RG 1.183 does not specifically address the WGDT Rupture event, the RG is utilized as a guide for performing the dose consequence analysis. The Fuel Handling Accident (FHA) release path for FHAs within the Fuel Handling Building (FHB), described in RG 1.183, Appendix B, most closely resembles that of the WGDT Rupture event. Therefore, this analysis is consistent with the guidance provided in RG 1.183, Appendix B. Regulatory Positions associated with the single WGDT Rupture event are discussed below:

1. Regulatory Position 4.1 – Radioactivity in the WGDT is assumed to be instantaneously released to the FHB and is subsequently released to the environment over a two hour period.
2. Regulatory Position 4.2 – The analysis does not assume a reduction in the amount of radioactivity released from the WGDT by the FHB ventilation system filters.
3. Regulatory Position 4.3 – The radioactivity released from the WGDT is assumed not to mix or dilute in the FHB.

Methodology:

The WGDTs receive radioactive gases from the liquids processed by the waste disposal system. The maximum storage of waste gases occurs after a refueling shutdown, at which time the WGDTs store the radioactive gases stripped from the reactor coolant. HBRSEP, Unit No. 2, TS 5.5.12 requires that the quantity of radioactive material contained in each WGDT be limited to less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area. The maximum noble gas activity in these tanks is administratively controlled by the HBRSEP, Unit No. 2, TRM, Specification 3.21, to 19,000 dose equivalent curies of Xe-133. The WGDTs are located in the FHB. In this event, a WGDT is assumed to rupture and the entire tank activity is assumed released instantaneously into the FHB.

The WGDT activity is assumed released within two hours (as in the FHA) from the FHB as a ground level release with no credit for building holdup, dilution, or air handling filters.

For this event, the Control Room ventilation system is assumed to remain in its normal mode at an intake rate of 400 cfm. No credit is taken for switchover to the emergency pressurization ventilation mode. Sensitivity analyses of the post-WGDT dose to Control Room operators are performed at assumed unfiltered inleakage rates of 85 cfm, 300 cfm, and 500 cfm. The sensitivity analyses show that lower inleakage assumptions produced higher Control Room dose consequences. During the postulated accident's first hour, a potential inleakage rate of 85 cfm is assumed, including an identified 70 cfm Hagan Room inleakage consistent with an Auxiliary Building exhaust fan failure. The analysis assumes that actions are taken to preclude inleakage from the Hagan Room at one hour. Therefore, an unfiltered inleakage of 15 cfm is assumed for the remainder of the event. This is the only accident evaluated for the AST where higher Control Room inleakage produces lower dose in the Control Room. Actions to preclude Hagan Room inleakage are not required to be implemented for this event because the higher inleakage from the Hagan Room will result in a lower Control Room dose.

Radiological Consequences:

The WGDT Rupture event does not assume a loss of offsite power. Therefore, any activity released during the event may be postulated to be released as a ground level release via the normal ventilation system, i.e., the plant stack. The point release-receptor locations are chosen to minimize the distance from the release point to the Control Room intake. The plant stack is closer to the Control Room than the nearest point on the FHB wall and is considered to be a more conservative release point. Therefore, the release is assumed from the plant stack.

For the EAB and LPZ dose calculations, the χ/Q factor for the zero to two hours time interval was assumed for all time periods. Using the zero to two hours χ/Q factor provides a more conservative determination of the EAB and LPZ dose, because the χ/Q factor for this time period is higher than for any other time period.

The radiological consequences of the design basis WGDT Rupture event are analyzed using the RADTRAD code and the input assumptions previously discussed. As shown in Attachment XIII, Table 2, "Dose Consequences," the radiological consequences of the WGDT Rupture event for the EAB, LPZ, and Control Room are within the appropriate acceptance criteria.

Environmental Qualification (EQ)

The HBRSEP, Unit No. 2, UFSAR, Section 3.11.5.2, discusses equipment EQ due to the radiation environment. RG 1.183, Regulatory Position 6, allows the licensee to use either the AST or the TID 14844 assumptions for performing the required EQ analyses until such time as a generic issue related to the effect of increased cesium releases on EQ doses is resolved. The HBRSEP, Unit No. 2, EQ analyses will continue to be based on the TID 14844 assumptions.

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

The current HBRSEP, Unit No.2, licensing commitments and plant design modifications in response to NUREG-0737, Items II.B.2, II.B.3, and II.F.1 are not affected by the incorporation of the AST.

The requirement to have and maintain the Post-Accident Sampling System (PASS) was eliminated by Amendment No. 192, dated January 14, 2002. Therefore, the incorporation of an AST does not affect this requirement.

Post-accident access to vital areas was ensured by a design review of plant shielding, as described in the UFSAR, Section 12.3.1.4. With the elimination of the PASS, vital area access is limited to the Control Room and the TSC/EOF. The dose consequences of the AST implementation on the Control Room and TSC/EOF have been provided in the analyses discussed in this submittal. The dose consequences for these areas were found to be within the regulatory acceptance criteria. Other areas of the plant are administratively controlled by Radiation Protection personnel to limit exposure.

Instrumentation potentially impacted by the implementation of the AST has been evaluated. This evaluation showed that the current TID 14844 based source term would, from a design standpoint, bound operation with the AST.

Technical Specifications Changes

The definition of Dose Equivalent I-131 in Section 1.1 is revised to reference NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation," December 1997, as the source of thyroid dose conversion factors. The dose consequences associated with the revised analyses incorporating the AST were determined using the dose conversion factors provided in NUREG/CR-6604, which are consistent with the International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, 1979. Therefore, changes to the definition of Dose Equivalent I-131 are consistent with the implementation of an AST.

The RCS operational leakage limits stated in LCO 3.4.13.d for total primary to secondary leakage through all SGs is reduced from 1 gpm to 0.3 gpm. In addition, the limit specified for primary to secondary leakage through any one SG (LCO 3.4.13.e) is reduced from 500 gallons per day to 150 gallons per day. The changes are made for consistency with the assumptions of those analyses that consider primary to secondary leakage through the SG tubes as an input (MSLB, SGTR, Locked Rotor, and Single RCCA Withdrawal). The changes are acceptable because they result in more conservative requirements and provide assurance that the input assumptions of the associated analyses are adequately protected.

The Dose Equivalent I-131 requirements of TS 3.4.16 are reduced from 1.0 μCi/gm to 0.25 μCi/gm in Condition A and in SR 3.4.16.2. The proposed value bounds the values for RCS equilibrium iodine assumed in the analyses. In addition, Figure 3.4.16-1 is deleted. Required Action A.1 is revised to replace the reference to the acceptable region of

Figure 3.4.16-1 with a limit of ≤ 60.0 μCi/gm. The second entry condition of Condition C is revised to replace the reference to the unacceptable region of Figure 3.4.16-1 with reference to > 60 μCi/gm. This change is made for consistency with the assumptions of those analyses that consider iodine spiking as an input (MSLB and SGTR). The associated analyses did not consider variable Dose Equivalent I-131 concentrations based on power level. Instead, the only concentration assumed for Dose Equivalent I-131 spiking was 60.0 μCi/gm. The changes are acceptable because they result in more conservative requirements and provide assurance that the input assumptions of the associated analyses are adequately protected.

TS 5.5.12.b is revised to delete "whole body" from the acceptance criteria. The WGDT Rupture event was reanalyzed due to changes in the atmospheric dispersion factors and to evaluate this event on the same basis as other offsite dose consequences. Therefore, the reference to whole body is deleted and a reference to TEDE has been added. This change acceptable because it provides assurance that the acceptance criteria provided in the TS is consistent with the criteria evaluated for the WGDT Rupture event.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Carolina Power & Light (CP&L) Company is proposing a change to the Appendix A, Technical Specifications (TS), of Facility Operating License No. DPR-23, for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. This change revises the licensing basis for HBRSEP, Unit No. 2, to implement the Alternative Source Term (AST) described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," in evaluating the radiological consequences of the following UFSAR Chapter 15 accidents:

- Main Steam Line Break,
- Reactor Coolant Pump Shaft Seizure (Locked Rotor),
- Single Rod Control Cluster Assembly (RCCA) Withdrawal,
- Steam Generator Tube Rupture (SGTR),
- Large Break Loss-of-Coolant Accident (LBLOCA), and
- Waste Gas Decay Tank (WGDT) Rupture

In addition, revised atmospheric dispersion factors for onsite and offsite dose consequences have been calculated and incorporated in the reanalysis of these events. As part of the full implementation of this AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11. A request for a selective implementation of an AST in the HBRSEP, Unit No. 2, Fuel Handling Accident analysis was previously submitted by letter dated March 13, 2002.

The full implementation of the AST is supported by the following Technical Specification (TS) changes:

> The definition of Dose Equivalent I-131 in Section 1.1 is revised to reference NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation," December 1997, as the source of thyroid dose conversion factors.
>
> The Reactor Coolant System (RCS) operational leakage limits, stated in Limiting Condition for Operation (LCO) 3.4.13, for total primary to secondary leakage through all steam generators is reduced from 1 gpm to 0.3 gpm. In addition, the limit specified for primary to secondary leakage through any one steam generator is reduced from 500 gallons per day to 150 gallons per day.
>
> The Dose Equivalent I-131 requirements of TS 3.4.16, "RCS Specific Activity," are reduced from 1.0 μCi/gm to 0.25 μCi/gm in Condition A and in Surveillance Requirement (SR) 3.4.16.2. In addition, Figure 3.4.16-1, "Reactor Coolant Dose Equivalent I-131 Specific Activity Limit Versus Percent of Rated Thermal Power," is deleted. Required Action A.1 is revised to replace the reference to the acceptable

region of Figure 3.4.16-1 with a limit of ≤ 60.0 μCi/gm. The second entry condition of Condition C is revised to replace the reference to the unacceptable region of Figure 3.4.16-1 with reference to > 60 μCi/gm.

The description of the Explosive Gas and Storage Tank Radioactivity Monitoring Program provided in TS 5.5.12 is revised to incorporate the TEDE as the acceptance criteria for dose consequences.

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

1. The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

   Implementation of the Alternative Source Term does not affect the design or operation of HBRSEP, Unit No. 2. Rather, once the occurrence of an accident has been postulated, the new source term is an input to evaluate the consequences of the postulated accident. The implementation of the Alternative Source Term has been evaluated in revisions to limiting design basis accidents at HBRSEP, Unit No. 2. Based on the results of these analyses, it has been demonstrated that, with the requested changes to the Technical Specifications, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC. This guidance is presented in 10 CFR 50.67 and Regulatory Guide 1.183. The proposed Technical Specifications changes result in more restrictive requirements and support the revisions to the limiting design basis accident analyses.

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

2. The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated.

   The proposed changes do not affect plant structures, systems or components. The Alternative Source Term and those plant systems affected by implementing the proposed changes do not initiate design basis accidents.

   Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

   The proposed changes are associated with the implementation of a new licensing basis for HBRSEP, Unit No. 2. The new licensing basis implements an Alternative Source

Term in accordance with 10 CFR 50.67 and the associated Regulatory Guide 1.183. The results of the revised limiting design basis analyses are subject to revised acceptance criteria. The analyses have been performed using conservative methodologies in accordance with the regulatory guidance. The dose consequences of the limiting design basis events are within the acceptance criteria found in the regulatory guidance associated with Alternative Source Terms.

The proposed changes continue to ensure that doses at the exclusion area and low population zone boundaries, as well as the Control Room, are within the corresponding regulatory limits. Specifically, the margin of safety for the radiological consequences of these accidents is considered to be that provided by meeting the applicable regulatory limits, which are conservatively set below the 10 CFR 50.67 limits. With respect to Control Room personnel doses, the margin of safety (the difference between the 10 CFR 50.67 limits and the regulatory limits defined by 10 CFR 50, Appendix A, Criterion 19 (GDC-19)) continues to be satisfied.

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

Based on the above discussion, CP&L has determined that the requested change does not involve a significant hazards consideration.

ENVIRONMENTAL IMPACT CONSIDERATION

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in an increase in individual or cumulative occupational radiation exposure. Carolina Power and Light (CP&L) Company has reviewed this request and determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is required in connection with the issuance of the amendment. The basis for this determination follows:

CP&L is proposing a change to the Appendix A, Technical Specifications (TS), of Facility Operating License No. DPR-23, for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. This change revises the licensing basis for HBRSEP, Unit No. 2, to implement the Alternative Source Term (AST) described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," in evaluating the radiological consequences of the following UFSAR Chapter 15 accidents:

- Main Steam Line Break,
- Reactor Coolant Pump Shaft Seizure (Locked Rotor),
- Single Rod Control Cluster Assembly (RCCA) Withdrawal,
- Steam Generator Tube Rupture (SGTR),
- Large Break Loss-of-Coolant Accident (LBLOCA), and
- Waste Gas Decay Tank (WGDT) Rupture

In addition, revised atmospheric dispersion factors for onsite and offsite dose consequences have been calculated and incorporated in the reanalysis of these events. As part of the full implementation of this AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11.

The full implementation of the AST is supported by the following Technical Specification (TS) changes:

> The definition of Dose Equivalent I-131 in Section 1.1 is revised to reference NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation," December 1997, as the source of thyroid dose conversion factors.

The Reactor Coolant System (RCS) operational leakage limits, stated in Limiting Condition for Operation (LCO) 3.4.13, for total primary to secondary leakage through all steam generators is reduced from 1 gpm to 0.3 gpm. In addition, the limit specified for primary to secondary leakage through any one steam generator is reduced from 500 gallons per day to 150 gallons per day.

The Dose Equivalent I-131 requirements of TS 3.4.16, "RCS Specific Activity," are reduced from 1.0 μCi/gm to 0.25 μCi/gm in Condition A and in Surveillance Requirement (SR) 3.4.16.2. In addition, Figure 3.4.16-1, "Reactor Coolant Dose Equivalent I-131 Specific Activity Limit Versus Percent of Rated Thermal Power," is deleted. Required Action A.1 is revised to replace the reference to the acceptable region of Figure 3.4.16-1 with a limit of ≤ 60.0 μCi/gm. The second entry condition of Condition C is revised to replace the reference to the unacceptable region of Figure 3.4.16-1 with reference to > 60 μCi/gm.

The description of the Explosive Gas and Storage Tank Radioactivity Monitoring Program provided in TS 5.5.12 is revised to incorporate the TEDE as the acceptance criteria for dose consequences.

Basis

The proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in the No Significant Hazards Consideration Determination, the proposed change does not involve a significant hazards consideration.

2. As demonstrated in the No Significant Hazards Consideration Determination, the proposed change does not result in a significant increase in the consequences of an accident previously evaluated and does not result in the possibility of a new or different kind of accident. Therefore, the proposed change does not result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite.

3. The Alternative Source Term does not affect the design or operation of the facility. Rather, once the occurrence of an accident has been postulated, the Alternative Source Term is an input to evaluate the consequences. The implementation of the Alternative Source Term has been evaluated in revisions to the analyses of the limiting design basis accidents at HBRSEP, Unit No. 2. Based on the results of these analyses, it has been demonstrated that, with the requested Technical Specifications changes, the dose consequences of the limiting event are within the regulatory guidance provided by the NRC for use with the Alternative Source Term. Therefore, the proposed change does not result in a significant increase in either individual or cumulative occupational radiation exposures.

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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

MARKUP OF TECHNICAL SPECIFICATIONS PAGES

1.1 Definitions (continued)

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL OPERATIONAL TEST (COT)

A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

NUREG/CR-6604 "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," December 1977. These dose conversion factors are consistent with ICRP-30.

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~Regulatory Guide 1.109, Rev. 1, NRC, 1977~~.

Ē – AVERAGE DISINTEGRATION ENERGY

Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

0.3

d. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and

150

e. 500 gallons per day primary to secondary LEAKAGE through any one SG.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

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

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm. 0.25	-----------Note----------- LCO 3.0.4 is not applicable. -------------------------		
	A.1	Verify DOSE EQUIVALENT I-131 ~~within the acceptable region of Figure 3.4.16-1.~~ ≤60.0 μCi/gm	Once per 4 hours
	AND		
	A.2	Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1	Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 ~~in the unacceptable region of Figure 3.4.16-1~~. >60.0 μCi/gm	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/Ē μCi/gm.	7 days
SR 3.4.16.2	------------------NOTE------------------ Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm. 0.25	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

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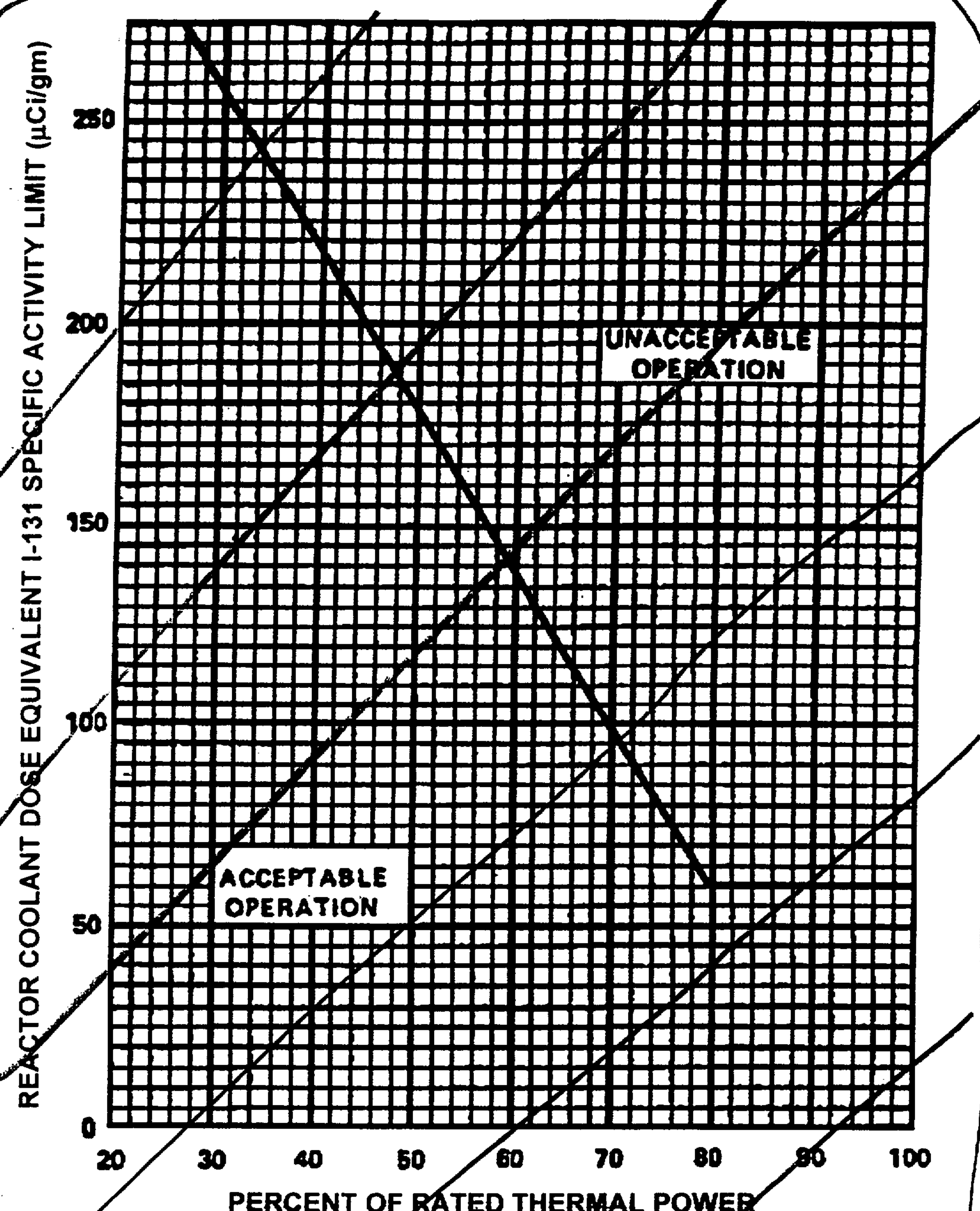

Figure 3.4.16-1
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit Versus Percent of RATED THERMAL POWER

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each Waste Gas Decay Tank is less (an) TEDE than the amount that would result in ~~a whole body~~ exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in each outdoor liquid radwaste tank that is not surrounded by liners, dikes, or walls, capable of holding the tank's contents and that does not have tank overflows and surrounding area drains connected to the Liquid Waste Disposal System is less than or equal to ten (10) Curies, excluding tritium and dissolved or entrained noble gases.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program shall be established requiring testing of both new fuel oil and stored fuel oil. The program shall include sampling and testing requirements, and acceptance criteria. The testing methods shall be in accordance with applicable ASTM Standards. The acceptance criteria shall be in accordance with the diesel engine manufacturer specifications. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has not become contaminated with other products during transit, thus altering the quality of the fuel oil.

(continued)

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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

RETYPED TECHNICAL SPECIFICATIONS PAGES

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," December 1977. These dose conversion factors are consistent with ICRP-30.
$\bar{E}$-AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

d. 0.3 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and

e. 150 gallons per day primary to secondary LEAKAGE through any one SG.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

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

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 0.25 μCi/gm.	------------Note------------ LCO 3.0.4 is not applicable. -----------------------------	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 60 μCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 > 60 μCi/gm.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/Ē μCi/gm.	7 days
SR 3.4.16.2	-----------------NOTE----------------- Only required to be performed in MODE 1. --------------------------------------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.25 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

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5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each Waste Gas Decay Tank is less than the amount that would result in an exposure of ≥ 0.5 rem TEDE to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in each outdoor liquid radwaste tank that is not surrounded by liners, dikes, or walls, capable of holding the tank's contents and that does not have tank overflows and surrounding area drains connected to the Liquid Waste Disposal System is less than or equal to ten (10) Curies, excluding tritium and dissolved or entrained noble gases.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program shall be established requiring testing of both new fuel oil and stored fuel oil. The program shall include sampling and testing requirements, and acceptance criteria. The testing methods shall be in accordance with applicable ASTM Standards. The acceptance criteria shall be in accordance with the diesel engine manufacturer specifications. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has not become contaminated with other products during transit, thus altering the quality of the fuel oil.

(continued)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

ATMOSPHERIC DISPERSION FACTORS

Table 1 – Offsite Atmospheric Dispersion Factors (χ/Q)

Time Period	**EAB**	**LPZ**
0 – 2 hours	1.77E-03	8.92E-05
2 – 8 hours	1.00E-03	3.50E-05
8 – 24 hours	7.58E-04	2.19E-05
1 – 4 days	4.11E-04	7.95E-06
4 – 30 days	1.71E-04	1.85E-06

Table 2 – Release-Receptor Combination Parameters

Release Point	Receptor Location	Straight Line Horizontal Distance(1)	Azimuth from Receptor Location to Release Point(2)	Receptor Location Elevation	Release Point Elevation
Plant Stack	Control Room	152.25 ft (46.41 m)	335°	250.58 ft	425 ft
MSSV/PORV	Control Room	141.07 ft (43 m)	289°	250.58 ft	275.58 ft
Closest Main Steam Line	Control Room	144.65 ft (44.1 m)	293°	250.58 ft	264.92 ft
Containment Nearest Point	Control Room	111.25 ft (33.91 m)	313°	250.58 ft	250.58 ft(3)
Containment Nearest Point	TSC/EOF	644 ft (196.3 m)	56°	236.67 ft	236.67 ft(3)
RHR Heat Exchanger Room	Control Room	87.12 ft (26.55 m)	336°	250.58 ft	250.58 ft(3)
RHR Heat Exchanger Room	TSC/EOF	705.8 ft (215.1 m)	56°	236.67 ft	236.67 ft(3)
FHB Wall	Control Room	209.69 ft (63.9 m)	335°	250.58 ft	250.58 ft(3)

(1) Straight line distance reduced by 5 ft for conservatism.
(2) Corrected to true north by adding 6°23' or 6.38° to plant north reference value and rounded up to nearest whole degree.
(3) Release elevation conservatively set equal to receptor elevation.

Table 3 – Onsite Atmospheric Dispersion Factors (χ/Q)
(Note: These factors are not corrected for occupancy)

Release-Receptor Combination	Control Room χ/Q (sec/m^3)				
	0-2 hours	2-8 hours	8-24 hours	1-4 days	4-30 days
Plant Stack-Control Room	1.23E-03	8.93E-04	3.60E-04	2.58E-04	2.19E-04
Closest MSSV/PORV-Control Room	2.59E-03	1.61E-03	7.11E-04	4.95E-04	4.04E-04
Closest Main Steam Line-Control Room	2.45E-03	1.54E-03	6.97E-04	4.72E-04	3.96E-04
Containment Nearest Point-Control Room	4.11E-03	2.68E-03	1.16E-03	8.14E-04	6.76E-04
Containment Nearest Point-TSC/EOF	1.64E-04	1.42E-04	6.50E-05	4.60E-05	3.52E-05
RHR Heat Exchanger Room-Control Room	7.12E-03	5.34E-03	2.26E-03	1.68E-03	1.36E-03
RHR Heat Exchanger Room -TSC/EOF	1.38E-04	1.22E-04	5.54E-05	4.01E-05	3.02E-05
FHB Wall-Control Room	1.33E-03	9.96E-04	4.25E-04	3.17E-04	2.56E-04

Table 4– Onsite Atmospheric Dispersion Factors (χ/Q)
(Note: These factors are corrected for occupancy)

Release-Receptor Combination	Control Room χ/Q (sec/m^3)				
	0-2 hours	**2-8 hours**	**8-24 hours**	**1-4 days**	**4-30 days**
Plant Stack-Control Room	1.23E-03	8.93E-04	3.60E-04	1.55E-04	8.76E-05
MSSV/PORV-Control Room	2.59E-03	1.61E-03	7.11E-04	2.97E-04	1.62E-04
Closest Main Steam Line-Control Room	2.45E-03	1.54E-03	6.97E-04	2.83E-04	1.58E-04
Containment Nearest Point-Control Room	4.11E-03	2.68E-03	1.16E-03	4.88E-04	2.70E-04
Containment Nearest Point-TSC/EOF	1.64E-04	1.42E-04	6.50E-05	2.76E-05	1.41E-05
RHR Heat Exchanger Room -Control Room	7.12E-03	5.34E-03	2.26E-03	1.01E-03	5.44E-04
RHR Heat Exchanger Room -TSC/EOF	1.38E-04	1.22E-04	5.54E-05	2.41E-05	1.21E-05
FHB Wall-Control Room	1.33E-03	9.96E-04	4.25E-04	1.90E-04	1.02E-04

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

FISSION PRODUCT INVENTORY

Table 1 – Core Activity Available for Release (LOCA)

Isotopes	Curies	Isotopes	Curies	Isotopes	Curies
Co-58	5.99E+05	Ru-103	9.87E+07	Cs-136	3.52E+06
Co-60	4.58E+05	Ru-105	6.83E+07	Cs-137	8.87E+06
Kr-85	7.30E+05	Ru-106	3.73E+07	Ba-139	1.15E+08
Kr-85m	1.51E+07	Rh-105	6.33E+07	Ba-140	1.13E+08
Kr-87	3.03E+07	Sb-127	5.38E+06	La-140	1.17E+08
Kr-88	4.20E+07	Sb-129	2.03E+07	La-141	1.02E+08
Rb-86	1.16E+05	Te-127	5.31E+06	La-142	9.83E+07
Sr-89	5.90E+07	Te-127m	8.87E+05	Ce-141	1.04E+08
Sr-90	6.16E+06	Te-129	1.90E+07	Ce-143	9.55E+07
Sr-91	7.39E+07	Te-129m	3.84E+06	Ce-144	8.18E+07
Sr-92	7.88E+07	Te-131m	1.23E+07	Pr-143	9.34E+07
Y-90	6.62E+06	Te-132	8.91E+07	Nd-147	4.17E+07
Y-91	7.69E+07	I-131	6.20E+07	Np-239	1.25E+09
Y-92	7.93E+07	I-132	9.02E+07	Pu-238	2.81E+06
Y-93	6.07E+07	I-133	1.28E+08	Pu-239	2.44E+04
Zr-95	1.05E+08	I-134	1.41E+08	Pu-240	3.55E+04
Zr-97	1.00E+08	I-135	1.21E+08	Pu-241	9.89E+06
Nb-95	1.06E+08	Xe-133	1.28E+08	Am-241	1.18E+04
Mo-99	1.16E+08	Xe-135	3.68E+07	Cm-242	3.23E+06
Tc-99m	1.03E+08	Cs-134	1.25E+07	Cm-244	3.88E+05

Table 2 - Core Activity Available for Release (Non-LOCA events)

Isotopes	Curies	Isotopes	Curies	Isotopes	Curies
Kr-85	7.30E+05	I-131	6.20E+07	Xe-133	1.28E+08
Kr-85m	1.51E+07	I-132	9.02E+07	Xe-135	3.68E+07
Kr-87	3.03E+07	I-133	1.28E+08	Cs-134	1.25E+07
Kr-88	4.20E+07	I-134	1.41E+08	Cs-136	3.52E+06
Rb-86	1.16E+05	I-135	1.21E+08	Cs-137	8.87E+06

Table 3 – Reactor Coolant System Equilibrium Activity Limited By 0.5 μCi/gm Dose Equivalent I-131 Prior To The Accident

Isotope	μCi/cc	Isotope	μCi/cc	Isotope	μCi/cc
Kr-85	1.08E+00	I-131	3.86E-01	Xe-133	4.27E+01
Kr-85m	2.60E-01	I-132	1.42E-01	Xe-135	1.18E+00
Kr-87	1.78E-01	I-133	6.22E-01	Cs-134	4.12E-02
Kr-88	6.40E-01	I-134	8.74E-02	Cs-136	5.92E-03
Rb-86	NEG.	I-135	3.34E-01	Cs-137	2.23E-01

Table 4 – Reactor Coolant System Equilibrium Activity Limited By 0.25 μCi/gm Dose Equivalent I-131 Prior To The Accident

Isotope	μCi/cc	Isotope	μCi/cc	Isotope	μCi/cc
Kr-85	5.41E-01	I-131	1.93E-01	Xe-133	2.14E+01
Kr-85m	1.30E-01	I-132	7.12E-02	Xe-135	5.88E-01
Kr-87	8.91E-02	I-133	3.11E-01	Cs-134	2.06E-02
Kr-88	3.20E-01	I-134	4.37E-02	Cs-136	2.96E-03
Rb-86	NEG.	I-135	1.67E-01	Cs-137	1.12E-01

Table 5 – Fraction of Fission Product Inventory Released During the Gap and Early In-Vessel Phases

Group	Isotopes	Gap Release Phase	Early In-Vessel Phase	Total
Noble Gases	Xe, Kr	0.05	0.95	1.00
Halogens	I, Br	0.05	0.35	0.40
Alkali Metals	Cs, Rb	0.05	0.25	0.30
Tellurium Metals	Te, Sb, Se	0.00	0.05	0.05
Ba, Sr	Ba, Sr	0.00	0.02	0.02
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co	0.00	0.0025	0.0025
Cerium Group	Ce, Pu, Np	0.00	0.0005	0.0005
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am	0.00	0.0002	0.0002

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

VENTILATION SYSTEM PARAMETERS

Table 1 – Control Room Ventilation System Parameters

Parameter	**Value**
Control Room Volume	20,124 ft^3
Normal Ventilation Flow Rates	
Filtered Makeup Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Makeup Flow Rate	400 cfm
Unfiltered Inleakage (Total) Non-LOCA (Except WGDT Rupture) LOCA	 300 cfm 170 cfm
Unfiltered Recirculation Flow Rate	N/A
Pressurization Mode Flow Rates	
Filtered Makeup Air Flow Rate	400 cfm
Filtered Recirculation Flow Rate	2600 cfm
Unfiltered Inleakage (Total) Non-LOCA (Except WGDT Rupture) LOCA	 300 cfm 170 cfm
Hagan Room Unfiltered Air Inleakage (Terminates after 1 hour)	70 cfm
Unfiltered Recirculation Flow Rate	N/A
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%

Table 2 – TSC/EOF Ventilation System Parameters

Parameter	Value
TSC/EOF Free Air Volume	262,640 ft^3
Normal Ventilation Flow Rates	
Unfiltered Makeup Flow Rate	3420 cfm
Unfiltered Inleakage (Total)	500 cfm
Pressurization Mode Flow Rates	
Filtered Makeup Air Flow Rate	3420 cfm
Filtered Recirculation Flow Rate	N/A
Unfiltered Inleakage (Total)	500 cfm
Filter Efficiencies	
Elemental	99%
Organic	99%
Particulate	99%

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

MAIN STEAM LINE BREAK

Table 1 – Analysis Inputs/Assumptions

Input/Assumption	Value
Power Level	2346 MWt (102% of current licensed power level)
Maximum Fuel Assembly/Batch Burnup	60,000 MWD/MTU
Maximum Linear Heat Generation Rate	6.12 kw/ft (for rod average burnups in excess of 54,000 MWD/MTU)
Fuel Enrichment	Bounding enrichment of 4.95 w/o (up to 5.0 w/o bounded by sensitivity study)
Maximum Radial Peaking Factor	1.8
Fuel Assemblies Total in Core Number Assumed to Breach (Failed Fuel Case)	 157 2
Core Fission Product Inventory (Failed Fuel Case)	Attachment VI, Table 2
RCS Equilibrium Activity	Attachment VI, Table 3
Release Fraction from Breached Fuel (Failed Fuel Case)	RG 1.183, Section 3.2, Table 3
Maximum Pre-Accident Spike Iodine Concentration	60 µCi/gm DE I-131
Maximum Equilibrium Iodine Concentration (Accident-Induced Spike and Failed Fuel Cases)	0.5 µCi/gm DE I-131
RCS Noble Gas and Alkali Metal Activity Prior to Accident	Proportional to 1% fuel defect. Activity level limited to proportion of proposed TS 3.4.16 limits.
RCS Halogen and Alkali Metal Partition Coefficient	100
Iodine Spike Appearance Rate	Attachment VIII, Table 3
Duration of Accident-Induced Spike	8 hours

Table 1 (continued)

Primary to Secondary Leakage Rate Faulted SG Unaffected SGs	 0.11 gpm 0.19 gpm
Secondary Coolant Iodine Activity Prior to Accident	0.1 μCi/gm DE I-131
Secondary Coolant Alkali Metal Activity Prior to Accident	10% of RCS concentration
SG Iodine and Alkali Metal Water/Steam Partition Coefficients	Affected SG - 1.0 Unaffected SG - 0.01
Time to Establish Shutdown Cooling and Terminate Release (Release Period)	53.2 hours to RHR in service 98.8 hours to leak termination
RCS Minimum Mass (Used for RCS Activity Calculation)	372,137 lbm
Iodine Chemical Form Released to Atmosphere from SGs	Particulate - 0% Elemental - 97% Organic - 3%
Secondary Coolant Mass Minimum (per SG) Maximum (per SG)	 88,641 lbm 137,294 lbm
Mass Releases to Environment and RADTRAD Flow Rates	Attachment VIII, Table 2
Atmospheric Dispersion Factors Offsite Onsite	 Attachment V, Table 1 Attachment V, Table 3 (Closest MSSV/PORV-Control Room Release-Receptor Combination)
Control Room Ventilation System Switch from Normal to Pressurization Mode	Attachment VII, Table 1 Automatic on SI signal in 50 seconds
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factors	RG 1.183, Section 4.2.6

Table 2 - MSLB Integrated Mass Releases

Time[1]	Steam Release from Affected SG (lbm)	Integrated Steam Release from Unaffected SGs (lbm)
0 hours	161,194[2]	0
0 - 2.0 hours	161,304.2[3]	300,116.1
2 - 8 hours	161,634.7[3]	861,350.9
8 - 24 hours	162,516.0[3]	1,971,677.3
24 - 53.2 hours	164,124.3[3]	3,582,768.8
53.2 - 98.8 hours	166,636.1[3]	N/A[4]

NOTES:
(1) Within time periods, flow rate assumed to be constant.
(2) Includes 23,900 lbm of Feedwater flow prior to isolation plus the 137,294 lbm SG steaming release. Auxiliary feedwater mass is not included because little activity is expected in the Auxiliary Feedwater system.
(3) Includes primary to secondary SG tube leakage.
(4) Steam release from unaffected SGs terminated at 53.2 hours.

Table 3 - Iodine Equilibrium Appearance Assumptions and Results

Parameter	Value
Inputs/Assumptions	
Maximum Nominal Letdown Flow	120 gpm at 130°F, 2235 psig
Uncertainty Applied to Letdown Flow	10%
Maximum Identified RCS Leakage	10 gpm
Maximum Unidentified RCS Leakage	1 gpm
Maximum RCS Volume	9623 ft^3 (433,859 lbm)
Results	
Isotopic Equilibrium Appearance Rates (Times 500)	
I-131	1.765E+06 μCi/sec
I-132	1.812E+06 μCi/sec
I-133	3.353E+06 μCi/sec
I-134	2.281E+06 μCi/sec
I-135	2.455E+06 μCi/sec

Table 4 – Dose Consequences

Item	EAB[1] (REM TEDE)	LPZ[2] (REM TEDE)	Control Room (REM TEDE)
Main Steam Line Break (Pre-Accident Iodine Spike)	0.26	0.03	0.14
Regulatory Limit (Pre-Accident Iodine Spike)	25[3]	25[3]	5[4]
Main Steam line Break (Accident-Induced Iodine Spike)	0.75	0.10	0.45
Regulatory Limit (Accident-Induced Iodine Spike)	2.5[3]	2.5[3]	5[4]
Main Steam line Break (Fuel Failures)	2.92	0.42	1.59
Regulatory Limit (Fuel Failures)	25[3]	25[3]	5[4]

NOTES:

(1) Worst two hour integrated dose.
(2) 30-day integrated dose.
(3) RG 1.183, Table 6, "Accident Dose Criteria."
(4) 10 CFR 50.67 and 10 CFR 50, Appendix A, Criterion 19.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR)

Table 1 – Analysis Inputs/Assumptions

Input/Assumption	**Value**
Power Level	2346 MWt (102% of current licensed power level)
Core Fission Product Inventory	Attachment VI, Table 2
RCS Equilibrium Activity	Attachment VI, Table 3
Release Fraction from Breached Fuel	RG 1.183, Section 3.2, Table 3
Maximum Fuel Assembly/Batch Burnup	60,000 MWD/MTU
Maximum Linear Heat Generation Rate	6.12 kw/ft (for rod average burnups in excess of 54,000 MWD/MTU)
Fuel Enrichment	Bounding enrichment of 4.95 w/o (up to 5.0 w/o bounded by sensitivity study)
Fuel Assemblies Total in Core Number Assumed to Breach	 157 17
Maximum Radial Peaking Factor	1.8
RCS Noble Gas and Alkali Metal Activity Prior to Accident	Proportional to 1% fuel defect. Activity level limited to proportion of proposed TS 3.4.16 limits.
RCS Halogen and Alkali Metal Partition Coefficient	100
RCS Minimum Mass (575.9°F)	372,137 lbm
Primary to Secondary Leakage Rate	0.3 gpm total through 3 steam generators
Time to Establish Shutdown Cooling and Terminate Release	53.2 hours
SG Minimum Mass (per SG)	88,641 lbm

Table 1 (continued)

Secondary Coolant Iodine Activity Prior to Accident	0.1 μCi/gm DE I-131
Secondary Coolant Alkali Metal Activity Prior to Accident	10% of RCS concentration
Mass Releases to Environment and RADTRAD Flow Rates	Attachment IX, Table 2
SG Iodine and Alkali Metal Water/Steam Partition Coefficient	0.01
Atmospheric Dispersion Factors Offsite Onsite	 Attachment V, Table 1 Attachment V, Table 3 (Closest MSSV/PORV-Control Room Release-Receptor Combination)
Control Room Ventilation System Switch from Normal to Pressurization Mode	Attachment VII, Table 1 Automatic on Control Room area radiation monitor signal
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factors	RG 1.183, Section 4.2.6

Table 2 –Mass Releases For RADTRAD Model

Time Period (Hours)	Mass Release Per Time Period (lbm)	Integrated Mass Release (lbm)
0-2	301,967.3	301,967.3
2-8	566,768.3	868,735.6
8-24	1,124,995.8	1,993,731.4
24-53.2	1,637,910.1	3,631,641.5

Table 3 – Dose Consequences

Item	EAB[1] (REM TEDE)	LPZ[2] (REM TEDE)	Control Room (REM TEDE)
Locked Rotor	2.24	0.211	0.847
Regulatory Limit	2.5[3]	2.5[3]	5[4]

NOTES:

(1) Worst two hour integrated dose.
(2) 30-day integrated dose.
(3) RG 1.183, Table 6, "Accident Dose Criteria."
(4) 10 CFR 50.67 and 10 CFR 50, Appendix A, Criterion 19.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

SINGLE ROD CONTROL CLUSTER ASSEMBLY (RCCA) WITHDRAWAL

Table 1 – Analysis Inputs/Assumptions

Input/Assumption	**Value**
Power Level	2346 MWt (102% of current licensed power level)
Core Fission Product Inventory	Attachment VI, Table 2
RCS Equilibrium Activity	Attachment VI, Table 3
Release Fraction from Breached Fuel	RG 1.183, Section 3.2, Table 3
Release Fraction from Fuel that Reaches or Exceeds the Initiation Temperature of Fuel Melt	RG 1.183, Section 3.2, Table 2
Maximum Fuel Assembly/Batch Burnup	60,000 MWD/MTU
Maximum Linear Heat Generation Rate	6.12 kw/ft (for rod average burnups in excess of 54,000 MWD/MTU)
Fuel Enrichment	Bounding enrichment of 4.95 w/o (up to 5.0 w/o bounded by sensitivity study)
Maximum Radial Peaking Factor	1.8
Fuel Assemblies Total in Core Number Assumed to Breach Number Assumed to Reach or Exceed the Initiation Temperature of Fuel Melt	 157 1 3
Halogen and Alkali Metal Partition Coefficient	100
Primary to Secondary Leakage Rate	0.3 gpm total through 3 steam generators
RCS Noble Gas and Alkali Metal Activity Prior to Accident	Proportional to 1% fuel defect. Activity level limited to proportion of proposed TS 3.4.16 limits.
Time to Establish Shutdown Cooling and Terminate Release	53.2 hours

Table 1 (continued)

RCS Minimum Mass (575.9°F)	372,137 lbm
SG Minimum Mass (per SG)	88,641 lbm
Mass Releases to Environment and RADTRAD Flow Rates	Attachment X, Table 2
Secondary Coolant Iodine Activity Prior to Accident	0.1 μCi/gm DE I-131
Secondary Coolant Alkali Metal Activity Prior to Accident	10% of RCS concentration
Atmospheric Dispersion Factors Offsite Onsite	 Attachment V, Table 1 Attachment V, Table 3 (Closest MSSV/PORV-Control Room Release-Receptor Combination)
Control Room Ventilation System Switch from Normal to Pressurization Mode	Attachment VII, Table 1 Automatic on Control Room area radiation monitor signal
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factors	RG 1.183, Section 4.2.6

Table 2 –Mass Releases For RADTRAD Model

Time Period (Hours)	Mass Release Per Time Period (lbm)	Integrated Mass Release (lbm)
0-2	301,967.3	301,967.3
2-8	566,768.3	868,735.6
8-24	1,124,995.8	1,993,731.4
24-53.2	1,637,910.1	3,631,641.5

Table 3 – Worst Two Hour Time Interval By Release Source

Release Source	Time (Hours)
RCS Halogen Inventory	39.8
RCS Noble Gas Inventory	0.0
RCS Alkali Metal Inventory	33.7
Secondary Coolant Halogen Inventory	0.0
Secondary Coolant Alkali Metal Inventory	0.0

Table 4 – Dose Consequences

Item	EAB(1) (REM TEDE)	LPZ(2) (REM TEDE)	Control Room (REM TEDE)
Single RCCA Withdrawal	1.76	0.237	0.745
Regulatory Limit	2.5(3)	2.5(3)	5(4)

NOTES:

(1) Worst two hour integrated dose.
(2) 30-day integrated dose.
(3) RG 1.183, Table 6, "Accident Dose Criteria."
(4) 10 CFR 50.67 and 10 CFR 50, Appendix A, Criterion 19.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

STEAM GENERATOR TUBE RUPTURE (SGTR)

Table 1 – Analysis Inputs/Assumptions

Input/Assumption	Value
Power Level	2346 MWt (102% of current licensed power level)
Core Fission Product Inventory	Attachment VI, Table 2
RCS Equilibrium Activity	Attachment VI, Table 4
Release Fraction from Breached Fuel	RG 1.183, Section 3.2, Table 3
Release Fraction from Fuel that Reaches or Exceeds the Initiation Temperature of Fuel Melt	RG 1.183, Section 3.2, Table 2
Maximum Fuel Assembly/Batch Burnup	60,000 MWD/MTU
Maximum Linear Heat Generation Rate	6.12 kw/ft (for rod average burnups in excess of 54,000 MWD/MTU)
Fuel Enrichment	Bounding enrichment of 4.95 w/o (up to 5.0 w/o bounded by sensitivity study)
Fuel Assemblies Total in core	 157
Maximum Pre-Accident Spike Iodine Concentration	60 μCi/gm DE I-131
Maximum Equilibrium Iodine Concentration	0.25 μCi/gm DE I-131
Iodine Spike Appearance Rate	335 times
Duration of Accident Initiated Spike	8 hours
Halogen and Alkali Metal Partition Coefficient	100
Primary to Secondary Leakage Rate Faulted SG Intact SGs (2)	 0.08 gpm 0.22 gpm (total)

Table 1 (continued)

RCS Noble Gas and Alkali Metal Activity Prior to Accident	Proportional to 1% fuel defect. Activity level limited to proportion of proposed TS 3.4.16 limits.
Time to Establish Shutdown Cooling and Terminate Release	53.2 hours
RCS Minimum Mass (575.9°F)	372,137 lbm
SG Minimum Mass (per SG)e	88,641 lbm
Integrated Mass Releases	Attachment XI, Table 2
Secondary Coolant Iodine Activity Prior to Accident	0.1 μCi/gm DE I-131
Secondary Coolant Alkali Metal Activity Prior to Accident	10% of RCS concentration
SG Iodine and Alkali Metal Water/Steam Partition Coefficients Affected SG (Flashed Steam) Affected SG (Non-flashed Steam) Unaffected SG	 1.0 0.01 0.01
Break Flow Flash Fraction	30.27%
Atmospheric Dispersion Factors Offsite Onsite	 Attachment V, Table 1 Attachment V, Table 3 (Closest MSSV/PORV-Control Room Release-Receptor Combination)
Control Room Ventilation System Switch from Normal to Pressurization Mode	Attachment VII, Table 1 Automatic on Control Room area radiation monitor signal at 310 seconds
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factors	RG 1.183, Section 4.2.6

Table 2 - SGTR Integrated Mass Releases

Time(1)	Break Flow in Ruptured SG (lbm)	Steam Release from Ruptured SG (lbm)	Integrated Steam Release From Unaffected SGs (lbm)
0 – 0.5 hours	131,000	95,500	104,640.7
0 - 2.0 hours	131,000	95,500	302,695.8
0 - 8 hours	N/A	N/A	871,641.4
0 - 24 hours	N/A	N/A	2,002,409.4
0 - 53.2 hours	N/A	N/A	3,650,872.3

NOTES:
(1) Within time periods, flow rate assumed to be constant.

Table 3 - Iodine Equilibrium Appearance Assumptions and Results

Parameter	Value
Inputs/Assumptions	
Maximum Nominal Letdown Flow	120 gpm at 130°F, 2235 psig
Uncertainty Applied to Letdown Flow	10%
Maximum Identified RCS Leakage	10 gpm
Maximum Unidentified RCS Leakage	1 gpm
Maximum RCS Mass	433,859 lbm
Results	
Isotopic Equilibrium Appearance Rates (Times 335)	
I-131	5.913E+05 μCi/sec
I-132	6.069E+05 μCi/sec
I-133	1.123E+06 μCi/sec
I-134	7.642E+05 μCi/sec
I-135	8.225E+05 μCi/sec

Table 4 – Dose Consequences

Item	EAB[1] (REM TEDE)	LPZ[2] (REM TEDE)	Control Room (REM TEDE)
SGTR – Pre-Accident Iodine Spike	23.87	1.21	4.48
Regulatory Limit (Pre-Accident Iodine Spike)	25[3]	25[3]	5[4]
SGTR – Coincident Iodine Spike	1.99	0.10	0.36
Regulatory Limit (Coincident Iodine Spike)	2.5[3]	2.5[3]	5[4]

NOTES:

(1) Worst two hour integrated dose.
(2) 30-day integrated dose.
(3) RG 1.183, Table 6, "Accident Dose Criteria."
(4) 10 CFR 50.67 and 10 CFR 50, Appendix A, Criterion 19.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

LARGE BREAK LOSS-OF-COOLANT ACCIDENT (LBLOCA)

Table 1 – Analysis Inputs/Assumptions

Input/Assumption	**Value**
RADIONUCLIDE RELEASE INPUTS	
Power Level	2346 MWt (102% of current licensed power level)
Maximum Fuel Assembly/Batch Burnup	60,000 MWD/MTU
Fuel Enrichment	Bounding enrichment of 4.95 w/o (up to 5.0 w/o bounded by sensitivity study)
Fuel Assemblies Total in Core	 157
Core Fission Product Inventory	Attachment VI, Table 1
RCS Equilibrium Activity	Attachment VI, Table 3
Containment Leakage Rate 0 to 24 hours After 24 hours	 0.1% (by weight)/day 0.05% (by weight)/day
LOCA Release Phase Timing and Duration	Attachment XII, Table 2
Core Inventory Release Fractions (Gap Release and Early In-Vessel Phases)	Attachment VI, Table 5
ESF Systems Leakage Rate (from 21 minutes to 30 days)	4 gph (Two times the TRM 3.23 limit)
ESF Leakage Flash Fraction	5.3%

Table 1 (continued)

RADIONUCLIDE TRANSPORT INPUTS	
Containment Ground Release Location	Containment nearest point to Control Room ventilation intake and containment nearest point to TSC/EOF ventilation intake
ESF Ground Release	Auxiliary Building RHR heat exchanger room nearest point to Control Room ventilation intake, and Auxiliary Building RHR heat exchanger room nearest point to TSC/EOF ventilation intake
Control Room Ventilation System Switch from Normal to Pressurization Mode	Attachment VII, Table 1 Automatic on SI signal in 35 seconds
TSC/EOF Ventilation Switch to Emergency Air Filtration Mode	Attachment VII, Table 2 2 hours
RADIONUCLIDE REMOVAL INPUTS	
Containment Natural Deposition	0.1/hour
Containment Spray Region Volume	1,018,434 ft^3
Containment Unsprayed Region Volume	940,092 ft^3
Flow Rate Between Sprayed and Unsprayed Volumes	130,000 cfm
Spray Removal Rates Elemental Iodine Time to Reach DF of 200 Particulate Iodine (Terminated at 167 minutes) Time to Reach DF of 50	 20/hour 2.46 hours 3.057/hour 20.8 hours
Spray Operation Initiation Time – RWST Suction Termination Time – RWST Suction Initiation Time – Containment Sump Suction Termination Time – Containment Sump Suction	 3 minutes 77 minutes 100 minutes 167 minutes
Control Room Ventilation Emergency Pressurization Mode Outside Air Intake and Filtered Recirculation Flow Rate and Filter Efficiencies	Attachment VII, Table 1

Table 1 (continued)

TSC/EOF Emergency Air Filtration Mode Outside Air Intake Flow Rate and Filter Efficiencies	Attachment VII, Table 2
PERSONNEL DOSE CONVERSION INPUTS	
Atmospheric Dispersion Factors Offsite Onsite	 Attachment V, Table 1 Attachment V, Table 3
Breathing Rates Offsite Control Room and TSC/EOF	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room and TSC/EOF Occupancy Factors	RG 1.183, Section 4.2.6

Table 2 – LOCA Release Phases

Phase	Onset	Duration
Gap Release	30 seconds	0.5 hours
Early In-Vessel	0.5 hours	1.3 hours

Table 3 – Dose Consequences

Item	EAB(1) (REM TEDE)	LPZ(2) (REM TEDE)	Control Room (REM TEDE)	TSC/EOF (REM TEDE)
LBLOCA	22.5	1.64	4.85	1.98
Regulatory Limit	25(3)	25(3)	5(4)	5(4)

NOTES:
(1) Worst two hour integrated dose.
(2) 30-day integrated dose.
(3) RG 1.183, Table 6, "Accident Dose Criteria."
(4) 10 CFR 50.67 and 10 CFR 50, Appendix A, Criterion 19.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

WASTE GAS DECAY TANK (WGDT) RUPTURE

Table 1 – Analysis Inputs/Assumptions

Input/Assumption	Value
Fission Product Inventory	19,000 curies of I-133 dose equivalent
Release Type	Radioactivity is assumed to be instantaneously released to the FHB, and is subsequently released to the environment over a two hour period via the plant stack as a ground level release
Atmospheric Dispersion Factors Offsite Onsite	 Attachment V, Table 1 Attachment V, Table 3 (Plant Stack-Control Room Release-Receptor Combination)
Control Room Ventilation System Control Room Volume Filtered Makeup Flow Rate Filtered Recirculation Flow Rate Unfiltered Makeup Flow Rate Unfiltered Inleakage (Total) Hagan Room Unfiltered Air Inleakage Filter Efficiencies	 20,124 ft^3 0 cfm 0 cfm 400 cfm 85 cfm 70 cfm (Terminates after 1 hour) None assumed
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3, 0 – 2 hours RG 1.183, Section 4.2.6
Control Room Occupancy Factors	RG 1.183, Section 4.2.6

Table 2 – Dose Consequences

Item	EAB[1] (REM TEDE)	LPZ[2] (REM TEDE)	Control Room (REM TEDE)
WGDT Rupture	0.19	0.01	0.00339
Regulatory Limit	0.5[3]	0.5[3]	5[4]

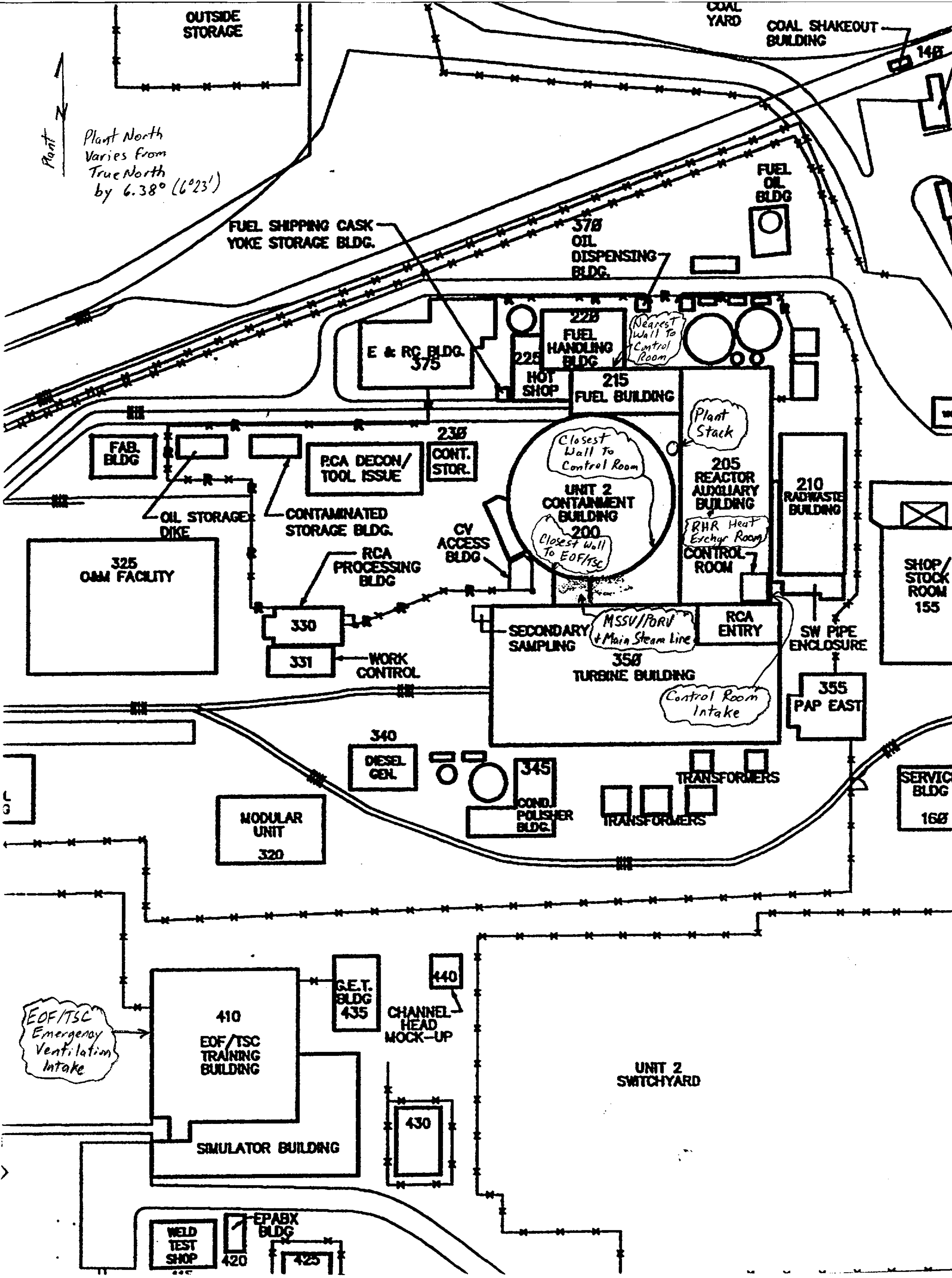
NOTES:

(1) Worst two hour integrated dose.
(2) 30-day integrated dose.
(3) HBRSEP, Unit No. 2, TS 5.5.12
(4) 10 CFR 50.67 and 10 CFR 50, Appendix A, Criterion 19.

United States Nuclear Regulatory Commission
Attachment XIV to Serial: RNP-RA/02-0067
2 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

ANNOTATED SITE DRAWING


OUTSIDE STORAGE
COAL YARD
COAL SHAKEOUT BUILDING
14Ø
Plant N
Plant North Varies from True North by 6.38° (6°23')
FUEL OIL BLDG
FUEL SHIPPING CASK YOKE STORAGE BLDG.
37Ø OIL DISPENSING BLDG.
E & RC BLDG. 375
225 HOT SHOP
22Ø FUEL HANDLING BLDG
Nearest Wall To Control Room
215 FUEL BUILDING
Plant Stack
FAB. BLDG
RCA DECON/ TOOL ISSUE
23Ø CONT. STOR.
Closest Wall To Control Room
UNIT 2 CONTAINMENT BUILDING 200
Closest Wall To EOF/TSC
205 REACTOR AUXILIARY BUILDING
210 RADWASTE BUILDING
OIL STORAGE DIKE
CONTAMINATED STORAGE BLDG.
CV ACCESS BLDG
RHR Heat Exchgr Room
CONTROL ROOM
325 O&M FACILITY
RCA PROCESSING BLDG
SHOP/ STOCK ROOM 155
330
SECONDARY SAMPLING
MSSV/PORV + Main Steam Line
RCA ENTRY
SW PIPE ENCLOSURE
331
WORK CONTROL
35Ø TURBINE BUILDING
Control Room Intake
355 PAP EAST
340 DIESEL GEN.
345 COND. POLISHER BLDG.
TRANSFORMERS
TRANSFORMERS
SERVICE BLDG 16Ø
MODULAR UNIT 320
G.E.T. BLDG 435
440
CHANNEL HEAD MOCK-UP
EOF/TSC Emergency Ventilation Intake
410 EOF/TSC TRAINING BUILDING
UNIT 2 SWITCHYARD
430
SIMULATOR BUILDING
WELD TEST SHOP
420
EPABX BLDG
425