CP&L
A Progress Energy Company

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BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ALTERNATIVE RADIOLOGICAL SOURCE TERM

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Parts 50.90 and 2.101, Carolina Power & Light (CP&L) Company is requesting a revision to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed license amendments are requested to support a full-scope application of an Alternative Source Term (AST) for BSEP Unit Nos. 1 and 2, with the exception that TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," continues to be used as the radiation dose basis for equipment qualification.

The AST analyses were performed following the guidance in Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors," dated July 2000, and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms."

The current operating license allows each BSEP unit to operate at a maximum power level of 2558 megawatts thermal (MWt). CP&L is currently pursing a project to increase the maximum licensed power level of each BSEP unit to 2923 MWt. In order to support the Extended Power Uprate project, the AST analyses have been performed at 102 percent of the uprated power level (i.e., 2981 MWt). No actual increase in licensed power levels is being sought by this submittal; a license amendment application for Extended Power Uprate is being submitted separately.

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Other changes to the current licensing bases for BSEP, Units 1 and 2, resulting from the Alternative Source Term analyses include:

- The AST analyses conservatively assume a five minute "positive pressure period" at the beginning of a LOCA event when drawdown of the secondary containment to a negative pressure occurs.

- Use of the Standby Liquid Control (SLC) system to buffer suppression pool pH is credited in preventing iodine re-evolution during a postulated radiological release as defined in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, Final Report," dated February 1, 1995. Credit for the SLC system in the AST analysis is based on operation of one SLC pump.

- Engineered Safety Feature and main steam isolation valve leakage are assumed to occur during post-LOCA conditions. The current licensing basis does not include unfiltered leakage (i.e., secondary containment bypass leakage).

- New offsite and Control Room atmospheric dispersion factors (χ/Qs) have been calculated using site-specific meteorology data collected between January 1996 and December 1999.

- The Technical Specifications are being revised to eliminate the requirement for the capability of automatic initiation of Control Room isolation during core alterations or fuel handling activities. This change will allow use of manual, instead of automatic, initiation of Control Room isolation following a fuel handling accident. Existing Technical Specification requirements for the automatic isolation of the Control Room following a LOCA, main steam line break, or control rod drop accident are being retained. The AST analyses have been performed assuming manual initiation of Control Room isolation at 20 minutes following a LOCA event or control rod drop accident. Therefore, CP&L plans to revise plant procedures to enforce the AST analysis assumption of manual initiation of Control Room isolation at 20 minutes following these accidents. For main steam line break events involving either isolation or non-isolation of the Control Room, the AST analyses demonstrate that the Control Room 30-day dose is bounding, regardless of Control Room isolation time.

The license amendment application makes use of an approach, previously accepted by the NRC, for evaluation of an alternate leakage path from the main steam isolation valves to the main condenser. As discussed in the Basis for Change Request accompanying this letter, CP&L is currently evaluating the results of plant seismic ruggedness inspections performed which support the alternate leakage path approach. These evaluations are not yet complete; however, CP&L has decided to proceed with submittal of this license amendment application, in order to allow NRC review of this license amendment application to begin, and to submit a supplement to the license amendment application regarding the alternate leakage path approach. CP&L plans to submit additional information regarding the alternate leakage path approach by September 28, 2001.

As stated in Regulatory Guide 1.183, the use of AST has no direct effect on the probability of the evaluated design basis accidents. The use of AST alone cannot increase the core damage frequency or the large early release frequency. Therefore, this license amendment application is not being submitted as a "risk-informed licensing action," as defined by Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," dated July 1998.

In letters dated February 15, 1997 (Serial: BSEP 97-0069), and February 28, 1997 (Serial: BSEP 97-0099), CP&L responded to NRC questions concerning assumptions and methodologies used in calculations of Control Room doses for a postulated main steam line break. A reactor coolant dose equivalent I-131 administrative limit of less than 0.1µCi/gram was implemented by CP&L until calculated results showed Control Room dose would be maintained less than General Design Criteria (GDC) 19 limits or until the NRC approved a license amendment application limiting I-131 peak activity. A license amendment application to limit I-131 peak activity was submitted in May 23, 1997 (Serial: BSEP 97-0222). In a letter dated April 17, 1998 (Serial: BSEP 98-0077), CP&L withdrew the license amendment application and submitted a calculation demonstrating that Control Room doses are below GDC 19 limits. However, CP&L committed to maintaining an administrative limit of 0.1µCi/gram dose equivalent I-131 until the NRC approved the calculation. By letter dated June 17, 1998, the NRC acknowledged this commitment. Since the April 17, 1998, submittal, numerous telephone conference discussions have been held with the NRC to discuss questions concerning the calculation. To date, the NRC has not issued a letter approving the calculation.

The AST analyses for a postulated main steam line break accident have been performed using the maximum reactor coolant concentrations allowed by the BSEP Technical Specifications (i.e., 4 µCi/gram and 0.2 µCi/gram dose equivalent I-131) rather than the administrative limits. The results of the AST analyses demonstrate that Control Room doses for a postulated main steam line break are well below the applicable regulatory limit. Therefore, effective immediately, CP&L is withdrawing its commitment to maintain an administrative limit for reactor coolant system dose equivalent I-131 activity.

The Alternative Source Term analyses have been performed without crediting secondary containment operability during fuel handling accidents. As such, the proposed license amendments relax operability requirements, during fuel handling and core alterations, for: (1) the Secondary Containment system, (2) Secondary Containment Isolation Instrumentation, (3) Secondary Containment Isolation Dampers, (4) the Standby Gas Treatment system, and (5) Control Room Emergency Ventilation system isolation instrumentation. These changes are consistent with Technical Specification Task Force (TSTF) 51, Revision 2, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," which was approved by the NRC on November 1, 1999. Since a portion of this license amendment application is based on TSTF-51, BSEP will adopt the commitment in TSTF-51 to follow NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," Section 4.5, for restoration capability for the secondary containment.

Revised BSEP, Unit 1 Bases pages associated with the proposed amendments are included in Enclosure 10. These pages are provided for information only and do not require issuance by the NRC.

Through a separate submittal, CP&L will be requesting approval of an Extended Power Uprate of the BSEP units. Issuance of the AST license amendments is required to support Extended Power Uprate.

CP&L requests issuance of the AST license amendment by February 28, 2002, to coincide with the Unit 1 Refueling Outage 13 (i.e., designated as B114R1). CP&L requests that the Unit 1 amendment, once approved, be issued with an implementation date of June 1, 2002.

For BSEP, Unit 2, CP&L requests that the Unit 2 amendment, once approved, be issued with an implementation date of March 1, 2003.

In accordance with 10 CFR 50.91(b), CP&L is providing a copy of this license amendment application to Mr. Mel Fry of the State of North Carolina. In accordance with 10 CFR 50.4(b)(1), CP&L is providing a copy of this license amendment application to the NRC Region II Office and the BSEP Resident Inspector.

Please refer any questions regarding this submittal to Mr. David C. DiCello, Manager - Regulatory Affairs, at (910) 457-2235.

Sincerely,

John S. Keenan

John S. Keenan

WRM/wrm

Enclosures:

1. Basis for Change Request
2. Alternative Source Term Safety Assessment
3. 10 CFR 50.92 Evaluation
4. Environmental Considerations
5. Page Change Instructions
6. Typed Technical Specification Pages - Unit 1
7. Typed Technical Specification Pages - Unit 2
8. Marked-up Technical Specification Pages - Unit 1
9. Marked-up Technical Specification Pages - Unit 2
10. Marked-up Technical Specification Bases Pages - Unit 1 (For Information Only)

John S. Keenan, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, and agents of Carolina Power & Light Company.

Dean S. Mason
Notary (Seal)

My commission expires: 8/29/04

cc (with enclosures):

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ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
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Basis For Change Request

Background

On December 23, 1999, in the Federal Register, the NRC published a new regulation, 10 CFR 50.67, providing a mechanism for licensed power reactors to replace the traditional accident source term used in design-basis accident analyses with alternative source terms (ASTs). Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (Reference 1). 10 CFR 50.67(b) states that licensees who seek to revise their current accident source term in design basis radiological consequence analyses should apply for a license amendment under 10 CFR 50.90.

In accordance with the Code of Federal Regulations, Title 10, Parts 50.90 and 2.101, Carolina Power & Light (CP&L) Company is requesting a revision to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, to support implementation of ASTs for BSEP. The AST analyses were performed using guidance contained in Regulatory Guide 1.183 and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms."

The current operating license allows each BSEP unit to operate at a maximum power level of 2558 megawatts thermal (MWt). However, CP&L is currently pursing a project to increase the maximum licensed power level of each BSEP unit to 2923 MWt. In order to support the Extended Power Uprate project, the AST analyses have been performed at 102 percent of the uprated power level (i.e., 2981 MWt). No actual increase in licensed power levels or change in fuel types is being sought by this submittal; a license amendment application for Extended Power Uprate is being submitted separately.

In addition to supporting implementation of ASTs, this license amendment application includes Technical Specification changes that partially incorporate a generic Boiling Water Reactor (BWR) Technical Specification change, Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," Revision 2. The TSTF-51 generic change has been previously approved by the NRC. TSTF-51 modifies Technical Specification requirements relating to core alterations and the handling of irradiated fuel in the secondary containment based on the recognition that, after reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. Based on the

AST analyses, certain Technical Specification requirements are being relaxed with respect to secondary containment, secondary containment isolation instrumentation, secondary containment isolation dampers, the Standby Gas Treatment (SGT) system, and Control Room Emergency Ventilation (CREV) system isolation instrumentation when core alterations are occurring or spent fuel is being moved.

The first portion of the discussion which follows summarizes the Technical Specification revisions and technical bases for the AST-related changes. The second portion of this discussion summarizes the Technical Specification revisions and technical bases for the TSTF-51 related changes.

Summary of AST-Related Technical Specification Changes

Implementation of the AST involves changes to the following Technical Specification:

Technical Specification	Affected Page(s) / Description
1.1 Definitions	Affected Page(s): 1.1-2 through 1.1-3 Technical Specification: The definition for DOSE EQUIVALENT I-131 is being revised to remove the word "thyroid" and to replace the reference to dose conversion factors from TID-14844 with a reference to Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989 and FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.

In conjunction with the Technical Specification change identified above, the AST analyses will result in the replacement of numerous references, in the Technical Specification Bases, to 10 CFR 100 with references to 10 CFR 50.67.

This license amendment application revises the BSEP, Unit 1 and 2 Technical Specifications to implement various assumptions in the AST analyses. The proposed license amendments provide for a change in design and licensing bases for full-scope application of the AST in accordance with 10 CFR 50.67, "Accident Source Term," to evaluations of the consequences of a design-basis loss-of-coolant accident (LOCA), main steam line break (MSLB) accident, control rod drop accident (CRDA), and fuel handling accident (FHA). The results of radiological analysis for these four limiting design-basis accidents are provided in Enclosure 2, "Alternative Source Term Safety Assessment."

Other events involving only reactor coolant releases (i.e., no fuel failures), such as the feedwater line break outside of containment are bounded by the releases associated with the MSLB outside containment.

The AST analyses were performed following the guidance in Regulatory Guide 1.183 and Standard Review Plan Section 15.0.1. Changes to the current licensing bases for BSEP, Unit Nos. 1 and 2, resulting from the AST analyses include:

- The AST analyses conservatively assume a five minute"positive pressure period" at the beginning of a LOCA event when drawdown of secondary containment to a negative pressure occurs.

- Use of the Standby Liquid Control (SLC) system to buffer suppression pool pH is credited in preventing iodine re-evolution during a postulated radiological release as defined in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, Final Report," dated February 1, 1995. Credit for the SLC system in the AST analysis is based on operation of one SLC pump.

- Engineered Safety Feature (ESF) and main steam isolation valve leakage are assumed to occur during post-LOCA conditions. The current licensing basis does not include unfiltered leakage (i.e., secondary containment bypass leakage).

- New offsite and Control Room atmospheric dispersion factors (χ/Qs) have been calculated using site-specific meteorology data collected between January 1996 and December 1999.

- The Technical Specifications are being revised to eliminate the requirement for the capability of automatic initiation of Control Room isolation during core alterations or fuel handling activities. This change will allow use of manual, instead of automatic, initiation of Control Room isolation following a fuel handling accident. Existing Technical Specification requirements for the automatic isolation of the Control Room following a LOCA, main steam line break, or control rod drop accident are being retained. The AST analyses have been performed assuming manual initiation of Control Room isolation at 20 minutes following a LOCA event or control rod drop accident. Therefore, CP&L plans to revise plant procedures to enforce the AST analysis assumption of manual initiation of Control Room isolation at 20 minutes following these accidents. For main steam line break events involving either isolation or non-isolation of the Control Room, the AST analyses demonstrate that the Control Room 30-day dose is bounding, regardless of Control Room isolation time.

Control Room Emergency Ventilation System

The current BSEP licensing basis assumes operation of the radiation protection mode (i.e., recirculation with filtered make-up) of the CREV system following a LOCA, main steam line break, control rod drop accident, or fuel handling accident. Following these postulated accidents, the CREV system automatically switches to the radiation protection mode of operation. For the LOCA and control rod drop accident, the AST analyses were performed conservatively assuming manual initiation of the Control Room radiation mode of operation at 20 minutes following the postulated accident. For the fuel handling accident, the AST analyses demonstrate that a 24-hour decay period is sufficient to ensure secondary containment and Control Room automatic isolation are not required during core alterations or fuel handling activities. Plant procedures will be revised to allow use of manual, instead of automatic, initiation of Control Room isolation following a fuel handling accident. For main steam line break events involving either isolation or non-isolation of the Control Room, the AST analyses demonstrate that the Control Room 30-day dose is bounding, regardless of Control Room isolation time.

Secondary Containment Drawdown

The AST analyses assume a five minute "positive pressure period" at the beginning of a LOCA event when drawdown of the secondary containment to a negative pressure occurs. The existing BSEP licensing basis does not require verification of a drawdown response as part of a periodic surveillance. Consequently, the time period assumed in the alternative source term analyses for drawdown has been determined based on a conservative analysis. The five minute drawdown period is 250 percent longer than the typical plant surveillance requirement in standard technical specifications (i.e., two minutes) for plants with similar SGT system designs. The current BSEP Surveillance Requirements (SR) will detect significant increases in secondary containment leakage (i.e., SR 3.6.4.1.3) or SGT system performance degradation (i.e., SR 3.6.4.3.1) prior to exceeding the assumed five minute drawdown period. Therefore, the existing BSEP Technical Specifications are adequate for ensuring this analysis assumption is met.

Iodine Re-evolution Methodology

Use of the SLC system to buffer suppression pool pH is credited in preventing iodine re-evolution during a postulated radiological release, as defined in NUREG-1465 and NUREG/CR-5950, "Iodine Evolution and pH Control," dated December 1992. The AST analyses assume operation of one SLC system pump to buffer suppression pool pH. The existing licensing basis for anticipated transients without scram (ATWS) assumes two SLC pump operation.

Atmospheric Dispersion (χ/Qs) Factors

Revised offsite atmospheric dispersion (χ/Qs) factors have been used in performing the AST analyses. The revised atmospheric dispersion factors were generated using PAVAN computer code. Revised Control Room atmospheric dispersion (χ/Qs) factors have been used in performing the AST analyses. The revised atmospheric dispersion factors were generated using ARCON96 computer code. Site-specific meteorological data collected for the time period from January 1996 to December 1999 has been used to calculate the revised atmospheric dispersion factors.

Engineered Safety Feature Leakage

ESF valve leakage has been assumed to occur during post-LOCA conditions. A leakage rate of 20 gallons per minute into the Reactor Building has been conservatively assumed in the AST analyses. This leakage rate is 20 times higher than the action limits established in existing plant programs and procedures.

Main Steam Isolation Valve Leakage

Main steam isolation valve (MSIV) leakage has also been assumed to occur during post-LOCA conditions. The AST analyses assume a total MSIV leakage rate of 46 scfh into the environment via the main steam lines and the condenser. As specified in SR 3.6.1.3.9, the maximum leakage rate allowed for each MSIV is 11.5 scfh (i.e., a total of 46 scfh for the four steam lines).

Regulatory Guide 1.183, Appendix A, provides assumptions acceptable to the NRC for evaluation of the radiological consequences of LOCAs using AST. For BWR MSIV leakage, Regulatory Guide 1.183 allows credit for a reduction in MSIV releases due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). Appendix A states that an acceptable model for evaluating reduction of MSIV releases is provided in General Electric Topical Report NEDC-31858P-A, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," September 1993.

In the Safety Evaluation for NEDC-31858P-A, Revision 2, (Reference 3), the NRC identified nine limitations to be addressed as part of a plant-specific application of the approach for evaluating MSIV leakage. These limitations relate to assuring that the alternate MSIV leakage path is functionally reliable commensurate with its intended safety function and assuring that the alternate leakage path, including the main condenser, is seismically rugged.

CP&L has performed a walkdown of BSEP, Unit 1, to assess the seismic ruggedness of the alternate leakage path. A similar walkdown for BSEP, Unit 2, is planned during Refueling Outage 15, which is currently scheduled to begin in March 2003. CP&L is using Earthquake Engineering, Incorporated (EQE) to evaluate the results of the BSEP, Unit 1, walkdowns. EQE is generally acknowledged as the industry experts in performing seismic ruggedness evaluations. The results of the BSEP, Unit 1, inspections are being evaluated to determine any modifications required to qualify this leakage path in accordance with NEDC-31858; however, these evaluations have not yet been completed. By September 28, 2001, CP&L plans to submit additional information in support of this license amendment application to address use of the alternate leakage path approach.

Equipment Environmental Qualification

Regulatory Guide 1.183, Section 1.3.5, provides discussion on the use of AST and environmental qualification analyses. Appendix I of the regulatory guide provides further guidance on the assumptions for evaluating radiation doses for equipment qualification (EQ).

Since the postulated increase in the post-accident integrated dose occurs only following an accident, there are no adverse affects on equipment relied upon to perform safety functions immediately following an accident. Appendix I of the regulatory guide states that the survivability period for any particular EQ component is the maximum duration, post-accident, that the component is expected to operate and perform its intended safety function.

Section 1.3.5 of the regulatory guide states that the NRC is assessing the effect of increased cesium releases on EQ doses to determine whether further action is warranted. Until this issue is resolved, the regulatory guide states that either the AST or the TID-14844 assumption may be used for performing EQ analyses. The regulatory guide further states that no plant modifications are required to address the impact of the difference in AST versus TID-14844 source term characteristics on EQ doses pending the outcome of the NRC evaluation of the generic issue.

Summary of TSTF-51, Revision 2 Related Technical Specification Changes

In addition to the Technical Specification changes associated with implementing the AST, this license amendment application includes changes that partially incorporate the generic BWR Technical Specification change TSTF-51, Revision 2. The TSTF-51 generic change has been previously approved by the NRC.

TSTF-51 modifies Technical Specification requirements relating to core alterations and the handling of irradiated fuel in the secondary containment based on the recognition that after reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. Based on the AST analyses, the proposed license amendments remove Technical Specification requirements for certain plant systems and equipment to be operable after sufficient radioactive decay has occurred to ensure offsite doses limits are not exceeded.

Implementation of TSTF-51-related changes involves the following Technical Specifications and Technical Specification Bases for both units:

Technical Specification	Affected Page(s) / Description
3.3.6.2 Secondary Containment Isolation Instrumentation	Affected Page(s): 3.3-62 (and Bases page B 3.3-179) Technical Specification: Footnote (b) of Table 3.3.6.2-1 is being revised to delete applicability of the associated function (i.e., Reactor Building Exhaust Radiation - High) during core alterations and require applicability only during movement of recently irradiated fuel assemblies. Bases: Remove references to the applicability for core alterations, incorporate applicability during movement of recently irradiated fuel assemblies, and describe the actual decay period required for recently irradiated fuel.
3.3.7.1 CREV System Instrumentation	3.3-63 (and Bases page B 3.3-187) Technical Specification: Deleting applicability of the specification during core alterations and requiring applicability only during movement of recently irradiated fuel assemblies. Bases: Remove references to the applicability for core alterations, incorporate applicability during movement of recently irradiated fuel assemblies, and describe the actual decay period required for recently irradiated fuel.

Technical Specification	Affected Page(s) / Description
3.6.4.1 Secondary Containment	Affected Page(s): 3.6-29 and 3.6-30 (and Bases page B 3.6-68 through 3.6-70) Technical Specification: Delete the Applicability requirement for core alterations and incorporate applicability during movement of recently irradiated fuel assemblies. Revise Condition C and its associated Required Actions to conform to those changes. Bases: Remove references to the applicability for core alterations, incorporate applicability during movement of recently irradiated fuel assemblies, and describe the actual decay period required for recently irradiated fuel.
3.6.4.2 Secondary Containment Isolation Dampers	Affected Page(s): 3.6-31 and 3.6-33 (and Base pages 3.6-73, -74, and -77). Technical Specification: Delete the Applicability requirement for core alterations and incorporate applicability during movement of recently irradiated fuel assemblies. Condition D and its associated Required Actions are being revised to conform to those changes. Bases: Remove references to the applicability for core alterations, incorporate applicability during movement of recently irradiated fuel assemblies, and describe the actual decay period required for recently irradiated fuel.

Technical Specification	Affected Page(s) / Description
3.6.4.3 SGT System	Affected Page(s): 3.6-34 through 3.6-36 (and Bases pages 3.6-80, -82, and -83). Technical Specification: Delete the Applicability requirement for core alterations and incorporate applicability during movement of recently irradiated fuel assemblies. Condition C, the Required Actions for Condition D, and Condition E and its associated Required Actions are being revised to conform to those changes. Bases: Remove references to the applicability for core alterations, incorporate applicability during movement of recently irradiated fuel assemblies, and describe the actual decay period required for recently irradiated fuel.

TSTF-51 Discussion

The Applicability statements for Technical Specifications 3.3.6.2, 3.6.4.1, 3.6.4.2, and 3.6.4.3 are being revised to remove the requirement that these Limiting Conditions of Operation be met during core alterations. As described in the Updated Safety Analysis Report (UFSAR), the accidents postulated to occur during core alterations are:

- Inadvertent criticality due to a control rod removal error or continuous control rod withdrawal error during refueling.
- The inadvertent loading and operation of a fuel assembly in an improper location.
- Fuel handling accidents.

The first two listed events are not postulated to result in fuel cladding integrity damage, and the fuel handling accident is the only event postulated to occur during core alterations that results in a significant radioactive release. The AST analyses have been performed assuming a ground release of all activity from the Reactor Building within 2 hours of the postulated accident, no holdup of release activity in the Reactor Building, no dilution of the release activity from the Reactor Building, and no operation of the SGT system following the postulated fuel handling accident. Therefore, based on these analysis assumptions, the Technical Specification requirements applicable to secondary containment operability, secondary containment isolation damper operability, SGT system operability, and isolation instrumentation operability for the CREV system during core alterations are being eliminated.

In addition, the AST analyses have been conservatively performed assuming manual isolation of the Control Room 20 minutes following the postulated fuel handling accident. The analyses also assume that release activity enters the Control Room at the normal ventilation rate during the time period from the initial fuel handling accident until the Control Room is isolated. This results in conservative dose assumptions for Control Room personnel. Therefore, based on these analysis assumptions, the Technical Specification requirements applicable to operability of the CREV system isolation instrumentation during core alterations are also being eliminated. Because the AST analyses take credit for operation of the CREV system beginning 20 minutes following the fuel handling accident, the TSTF-51 changes related to operability, during core alterations, of the CREV system (Technical Specification 3.7.3), Control Room Air Conditioning (AC) system (Technical Specification 3.7.4), AC Sources - Shutdown (Technical Specification 3.8.2), DC Sources - Shutdown (Technical Specification 3.8.5), and Distribution systems - Shutdown (Technical Specification 3.8.8) are not being requested.

For the Technical Specifications are that being revised to incorporate TSTF-51, the Applicability statements are being changed to include operations involving the movement of "recently irradiated fuel assemblies." Also, the wording of both the Conditions and Required Actions for these Technical Specifications are being modified consistent with the Applicability statement changes. The "recently irradiated fuel assemblies" terminology refers to irradiated fuel that contains sufficient fission products to require operability of accident mitigation systems for meeting accident analysis assumptions. The new terminology is being used to define the conditions where fuel handling activities can involve situations for which significant radioactive releases can be postulated. Thus, the new terminology is being used to modify the operability requirements for the identified safety systems. The AST analyses demonstrate that a 24-hour decay period is sufficient to ensure secondary containment and CREV system automatic isolation are not required to mitigate a FHA. The 24-hour decay period will be included in the Bases for each of the modified Technical Specifications.

The Applicability statements related to operations with a potential for draining the reactor vessel are unaffected by the proposed changes.

Supplemental Risk Discussion – Shutdown Controls

The following discussion of shutdown risk is provided to supplement the analysis and justification of the changes to relax the operational constraints during shutdown. It is applicable primarily to those Technical Specifications affected by the proposed changes regarding the terminology "recently irradiated fuel assemblies."

The containment and associated ESF systems are only required by the Technical Specifications to respond to the specific events which are postulated to result in a significant release of radioactivity (e.g., a fuel handling accident or a reactor cavity or fuel pool drain down). As a result, the requirements of the Technical Specifications are based on the plant being in specified conditions and are not based on providing requirements associated with shutdown risk considerations. Shutdown risk issues are instead addressed by utility outage management programs that follow the guidance of NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management." NUMARC 91-06, Section 4.5, discusses the need to assure that secondary containment closure can be achieved to prevent fission product release during severe accidents. NUMARC 91-06 also identifies that the time to effect closure should be consistent with plant conditions (e.g., reactor coolant system inventory and decay heat load). Consistent with the industry's commitment in the letter from NUMARC's President Mr. Byron Lee, Jr., to Mr. James M. Taylor of the NRC, BSEP will adopt the commitment in TSTF-51 to follow NUMARC 91-06, Section 4.5, for restoration capability for the secondary containment.

In the draft NUMARC 93-01 guideline, Section 11.2.6.5, "Safety Assessment for Removal of Equipment from Service During Shutdown Conditions," under the subheading of "Containment – Primary (PWR) / Secondary (BWR)," the following guidance is provided:

> ... for plants which obtain amendments to modify Technical Specification requirements on primary or secondary containment operability and ventilation system operability during fuel handling or core alterations, the following guidelines should be included in the assessment of systems removed from service:
>
> - During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.
>
> - A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt measures need not completely block the penetration or be capable of resisting pressure.

The purpose of the "prompt methods" mentioned above is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.

In the interim period until the revision of NUMARC 93-01 is endorsed as a formal industry position, BSEP will adopt provisions for controlling the removal from service of systems, structures, and components (SSCs) that are currently required by Technical Specifications during core alterations and irradiated fuel handling activities.

The proposed Technical Specification changes do not affect the requirements to have the containment systems operable any time the unit is in Mode 1, 2, or 3, regardless of whether irradiated fuel handling is occurring in the spent fuel pool.

This change does not impact the BSEP Outage Risk Assessment and Management (ORAM) computer-based Risk Assessment Program calculations of risk metrics (i.e., core damage risk and boiling risk). ORAM does not calculate the Large Early Release Frequency (LERF) risk profile. Of those accidents during Modes 4 and 5 which are postulated to result in a release, the fuel handling accident produces a small release and the loss of shutdown cooling event is a much more slowly evolving scenario that allows evacuation prior to release. Therefore, the LERF profile during this operation is essentially zero.

References

1. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.
2. NEDC-31858P-A, Volume 1, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems, Volume 1" dated August 1999.
3. Letter from Frank M. Akstulewicz (NRC) to T. A. Green, "Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, 'BWROG Report For Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems,' September 1993."

ENCLOSURE 2

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ALTERNATIVE RADIOLOGICAL SOURCE TERM

Alternative Source Term Safety Assessment

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ACRONYMS AND ABBREVIATIONS

AL	Analytical Limits
AST	Alternate Source Term
BSEP	Brunswick Steam Electric Plant
BWROG	Boiling Water Reactors Owners' Group
CRDA	Control Rod Drop Accident
CSCS	Core Standby Cooling Systems
DBA	Design Basis Accident
DEB	Double Ended Break
EAB	Exclusion Area Boundary
EDG	Emergency Diesel Generator
EOF	Emergency Operations Facility
EPU	Extended Power Uprate
EQ	Equipment Qualification
ESF	Engineered Safety Features
FHA	Fuel Handling Accident
GE	GE Nuclear Energy
HEPA	High Efficiency Particulate Air
HPCI	High Pressure Core Injection
LOCA	Loss of Coolant Accident
LPU	License Power Uprate
LPZ	Low Population Zone
MSIV	Main Steam Isolation Valve
MSLBA	Main Steam Line Break Accident
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OL	Operating License
PASS	Post Accident Sampling System
PPP	Positive Pressure Period
RCIC	Reactor Core Injection Cooling
RPV	Reactor Pressure Vessel
SGT	Standby Gas Treatment System
SCB	Secondary Containment Bypass
SLCS	Standby Liquid Control System
SRP	Standard Review Plan
TEDE	Total Effective Dose Equivalent
TS	Technical Specifications
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report

1. INTRODUCTION

1.1 Overview and Objective

The objective of this task is to perform BSEP Design Basis Accident (DBA) radiological consequence analyses to support implementation at the Brunswick Steam Electric Plant (BSEP), Unit, Nos. 1 and 2 of Alternate Source Terms (AST) described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 3). The AST analyses have been performed using the guidance in NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors" (Reference 5) and Standard Review Plan Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" (Reference 4).

Carolina Power & Light (CP&L) Company has performed analyses to support a full scope implementation of the AST, as defined in Section 1.2.1 of Regulatory Guide 1.183. The implementation consists of the following steps:

1. Identification of the alterative source term based on plant-specific analysis of the fission product inventory,
2. Calculation of the release fractions for the four boiling water reactor design basis accidents (DBAs),
3. Analysis of the atmospheric dispersion for the radiological propagation pathways,
4. Calculation of deposition and removal mechanisms,
5. Calculation of offsite, control room, and emergency response facility personnel Total Effective Dose Equivalent (TEDE) doses, and
6. Evaluation of other design and licensing bases, such as NUREG-0737, Item II.B.2, and Equipment Qualification.

The BSEP operating license currently limits operation to a maximum power level of 2558 megawatts thermal (MWt). Carolina Power & Light (CP&L) Company is currently pursuing a project to increase the maximum licensed power level of each BSEP unit to 2923 MWt. In support of the Extended Power Uprate (EPU) project, the AST analyses have been performed assuming operation at the Extended Power Uprate conditions (i.e., 2923 MWt and 1045 psia reactor steam dome pressure).

1.2. Summary

The radiological consequences of the plant's DBA accidents were analyzed for impact from EPU and AST implementation. Implementation of the AST as the plant radiological consequence analyses licensing basis requires a license amendment per the requirements of 10 CFR 50.67. The analyses demonstrate the offsite, Control Room, and Emergency Operations Facility (EOF)/Technical Support Center (TSC) post-accident doses remain under regulatory limits. An evaluation of the postulated post accident missions for compliance to NUREG-0737, Item II.B.2, demonstrates that the plant design is adequate to maintain the post-accident vital area access considering the impact from the EPU and AST.

2. EVALUATION

2.1 Scope

2.1.1 Accident Radiological Consequence Analyses

The following accident analyses documented in the BSEP Updated Final Safety Analysis Report (UFSAR) were addressed using methods and input assumptions consistent with the AST:

- UFSAR Section 15.4.6, Control Rod Drop Accident (CRDA)
- UFSAR Section 15.6.3, Main Steam Line Break Accident (MSLBA)
- UFSAR Section 15.6.4, Loss of Coolant Accident (LOCA), including Control Room habitability analysis discussed in UFSAR Sections 6.4.
- UFSAR Section 15.7.1, Refueling Accident

The analyses were based on operating conditions consistent with the EPU. In addition, the analysis assumptions follow the guidelines prescribed in Standard Review Plan (SRP) 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms" (Reference 4) and Regulatory Guide 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 5). The radiological release timing methodology also incorporated the results of the BWROG Report "Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR" (Reference 6).

The analysis results were evaluated to demonstrate compliance with the acceptance criteria presented in 10 CFR 50.67 and GDC 19 of 10 CFR 50 Appendix A.

UFSAR Section 15.6.4.5 addresses the impact of the TID-14844 source term on ESF systems, specifically, the Standby Gas Treatment (SGT) system performance and post-accident EQ impact on Core Standby Cooling system (CSCS) components. The EPU/AST implementation impact on the SGT performance was evaluated. The Control Room Ventilation System filters and the EOF/TSC Ventilation System filters were also evaluated for EPU/AST implementation. The post-accident EQ impact on CSCS components has been evaluated for EPU implementation using the TID-14844 source term, scaled for power, to account for the higher power level.

2.1.2 NUREG-0737, Item II.B.2

This portion of the task reviewed the current design basis for NUREG-0737, Item II.B.2, post-accident vital area access and determined the impact from EPU and AST implementation. Calculations used in support of NUREG-0737, Item II.B.2, shield design and vital area access mission dose assessments were included in the review. Mission doses were evaluated to determine that the current design basis contains sufficient margin to encompass the EPU and AST and continue to meet the dose criteria of NUREG-0737, Item II.B.2.

2.2 Method of Evaluation

2.2.1 Accident Radiological Consequence Analyses

New calculations were prepared for the simulation of the radionuclide release, transport, removal, and dose estimates associated with the plant's postulated accidents as listed in Section 2.1.1.

The MicroShield code, version 5.05 (Reference 10), was used in this task. MicroShield is a point kernel integration code used for general purpose gamma shielding analysis. This version of the code has been verified and validated in accordance with BSEP Standard Procedure EGR-NGGC-0016, "Engineering Analysis Software – Dedication and Benchmark Requirements." Although not considered an NRC approved code, MicroShield is used in safety related applications by many nuclear plants in the United States, including BSEP. The code has been used to support licensing submittals that have been accepted by the NRC.

The RADTRAD computer code (Reference 7), version 3.02, was used in this task. RADTRAD is a radiological consequence analysis code used to estimate post-accident doses at plant offsite locations and in the control room. This version of the code has been verified and validated in accordance with BSEP Standard Procedure EGR-NGGC-0016. The code is used by the NRC staff in safety reviews.

Offsite atmospheric dispersion factors (χ/Q's) were calculated with the PAVAN computer code (Reference 9). The PAVAN code calculates the diffusion from a source and relative concentration at a receiver due to an accidental release of radioactivity into the environment per the guidance in NRC Regulatory Guide 1.145 (Reference 14). The

PAVAN code has been verified and validated in accordance with BSEP Standard Procedure EGR-NGGC-0016. The code is used by the NRC staff in safety reviews.

Control room atmospheric dispersion factors (χ/Q's) were calculated with the ARCON96 computer code (Reference 8). The ARCON96 code calculates relative concentrations in plumes from nuclear power plants at control room air intakes in the vicinity of the release point. The ARCON96 code has been verified and validated in accordance with BSEP Standard Procedure EGR-NGGC-0016. The code is used by the NRC staff in safety reviews.

In addition to the calculational tools described above, the radiological consequence analyses made use of hand calculations and spreadsheets, supported by appropriate references, to determine inputs and outputs such as plant specific source terms, filter loading determinations, and suppression pool pH analyses.

2.2.2 NUREG-0737, Item II.B.2

Plant calculations used in support of the plant post-accident vital area access in accordance with the requirements of NUREG-0737, Item II.B.2, were reviewed. Post-accident personnel missions resulting in the bounding mission doses were identified. These mission doses were then reviewed for impact from the EPU and AST. Calculations were prepared to document the EPU and AST impact.

The MicroShield code, version 5.05 (Reference 10), was used in this task. MicroShield is a point kernel integration code used for general purpose gamma shielding analysis. This version of the code has been verified and validated in accordance with BSEP Standard Procedure EGR-NGGC-0016.

2.3 Inputs and Assumptions

2.3.1 Accident Radiological Consequence Analyses

All analyses were performed at 2981 MWt (102% of 2923 MWt) in accordance with NRC Regulatory Guide 1.49 (Reference 12).

2.3.1.1 LOCA Inputs and Assumptions

The key inputs used in this analysis are included in Tables 2-1 through 2-3. These inputs and assumptions fall into three main categories, Radionuclide Release Inputs, Radionuclide Transport Inputs, and Radionuclide Removal Inputs.

2.3.1.1.1 LOCA Release Inputs

The reactor core inventory used in the LOCA analysis was developed for EPU conditions using the ORIGEN code. The reactor core inventory for the LOCA analysis was based on an assumed fuel irradiation time, which develops "equilibrium" activities in the fuel (i.e., typically three years). Most fission products reach equilibrium within a three-year period. The calculated short-term inventories are approximately proportional to core thermal power. Consequently, for the EPU, the inventories of those isotopes, which reached or approached equilibrium, can conservatively be assumed to increase in proportion to the thermal power increase. The inventories of the very long-lived isotopes, which did not approach equilibrium, can also be assumed to increase proportionally if the fuel irradiation time remains within the original basis.

The planned 24-month equilibrium cycle for operation at EPU Rated Thermal Power (RTP) results in a bounded average irradiation time of about 1241 EFPDs (Effective Full Power Days), corresponding to 36 GWd/MT exposure, for the fuel bundles in core at end-of-cycle and a bounded average irradiation time of about 1724 EFPDs, corresponding to 50 GWd/MT exposure, for the discharged fuel bundles. The core fission product, transuranic nuclide and activation product inventories for the equilibrium core post-accident evaluations at 36 GWd/MT (end of cycle) and 50 GWd/MT (discharge) bound the corresponding source term inventories of the current and the transition cores up to the equilibrium cycle. The fuel bundles are assumed to have a bounded initial average enrichment of 4.5 weight percent. The fission product inventories were developed and defined on the basis of Curies per MWt at 50 GWd/MT for the LOCA and MSLB analyses and at 36 GWd/MT for the FHA and CRDA analyses. Therefore, the fission product inventories, used for post-accident evaluations, increase in proportion to reactor power.

The release source term is developed using the 60 isotope subset recommended by the NRC (in the RADTRAD code, Reference 7, BWR default inventory) and the guidance in NUREG-1465 and Regulatory Guide 1.183 (References 3 and 5) for release fractions, form, and timing as well as the BWROG gap release timing criteria specified in

References 5 and 6. This postulated LOCA source term represents a change in the BSEP design and licensing bases for radiological consequence analysis.

Per plant Technical Specifications, a 0.5%/day primary to secondary containment leakage rate (Reference 22, Item B.14) is assumed. In accordance with Appendix A, Section 3.7, of Reference 5, the primary to secondary leakage rate is reduced by 20% at 24 hours into the LOCA to 0.4%/day based on the post-LOCA drywell pressure history. A portion of the primary containment leakage, 23 scfh (Reference 22, Item B.20), is assumed to bypass the secondary containment and released to the atmosphere via the condenser. This secondary containment bypass leakage consists of 10 scfh via the inboard MSIV drain to the condenser, 10 scfh via the HPCI system drain to condenser and 3 scfh via the RCIC system drain to the condenser.

ESF systems are assumed to leak at a rate of 20 gpm into the Reactor Building which is, conservatively, 20 times higher as compared to the plant leakage "action" (identification and isolation) limits (Reference 22, Item B.26). The conservatism allows for a factor of 2 as required per Regulatory Guide 1.183 plus an additional safety factor of 10. The ESF leakage rate is assumed throughout the 30-day duration of the postulated accident.

A plant Technical Specifications total allowable drywell MSIV leakage of 46 scfh (Reference 22, Item B.35) is assumed to leak directly into the environment via the main steam lines and the turbine/condenser for the duration of the accident.

The NUREG-1465 and Regulatory Guide 1.183 (References 3 and 5) accident isotopic release specification allows deposition of iodine in the suppression pool. The iodine is assumed to remain in solution as long as the pool pH is maintained above 7. BSEP emergency operating procedures will be revised to direct operators, upon detection of symptoms indicating that core damage is occurring, to manually initiate the Standby Liquid Control (SLC) system. No credit is taken for any operator action during the first 10 minutes of an event. If an accident were to occur which would create the conditions assumed in the analyses, it is reasonable to assume that manual initiation of SLC injection would be initiated promptly. For the purposes of this analysis, however, this action is delayed for approximately 59 minutes after the event starts (or essentially within 57 minutes following the onset of fission product release) releasing the system's 2530 gallon inventory (Reference 22, Item G.13) of sodium pentaborate solution into the RPV within 2 hours after the postulated accident. In Calculation BNP-RAD-003, the buffering effect of the SLC system solution is shown to maintain the suppression pool pH above 7 for the 30-day duration of the postulated LOCA (Reference 18)) and prevent iodine re-evolution. As indicated above, BSEP emergency operating procedures direct operators to manually initiate the SLC system when core damage is believed to be occurring. Crediting this system, as an assumption in support of radiological consequence analysis, represents a change to the BSEP design and licensing basis.

2.3.1.1.2 LOCA Transport Inputs

At the beginning of the event, a loss of offsite power is assumed which results in the loss of reactor building ventilation that maintains secondary containment at a negative pressure with respect to the outside atmosphere. A Positive Pressure Period (PPP) was conservatively assumed to occur at time zero resulting in leakage from the secondary containment to the environment. The PPP was conservatively assumed to be terminated at 5 minutes, which includes time delays for Emergency Diesel Generators (EDG) start and the SGT system start and pressure drawdown. This 5-minute PPP was chosen to be sufficiently conservative such that it would bound actual system performance. Addition of this assumption to the current basis is necessary for compliance with the intent of the requirements of Appendix A of Regulatory Guide 1.183 (Reference 5). During the PPP, both the Containment and ESF leakage are assumed to be released directly to the environment, unfiltered, at ground level. To maximize the calculated post-accident doses, the ground level Reactor Building releases during the PPP were assumed to discharge from the building's closest location to the receptor location at the Control Room or the EOF/TSC air intake. The most conservative release location from either BSEP unit (Unit 1 Reactor Building South Wall and Unit 2 Reactor Building North Wall for the Control Room; Unit 2 Reactor Building South Wall for the EOF/TSC) was used as a representative release point from the Reactor Building throughout the release period. Following the PPP, all Reactor Building releases discharge from the SGT via the main stack.

The MSIV leak pathway was assumed to discharge from the low-pressure turbine at the Turbine Building's operating floor (Reference 22, Items B.51 and A.1.17 through A.1.20). This assumption implies dispersion of the activity at this "average" location corresponding to the building's available openings with no credit for building hold-up.

Atmospheric dispersion coefficients were calculated, for each identified release path, based on site specific meteorology data collected between January 1996 and

December 1999 (Reference 22, Section A.1.1) as documented in Calculations BNP-RAD-001 and BNP-RAD-002 (References 16 and 17).

The Control Room is automatically isolated and placed in the emergency ventilation mode upon an air intake radiation monitor isolation signal. For analysis purposes, the accident activity was allowed to enter the Control Room for 20 minutes into the LOCA at a conservative normal ventilation flow rate of 2100 cfm at which time the Control Room was assumed to be isolated by operator action (Reference 22, Items B.57, B.60, D.31, and D.34). Per Reference 22 Items B.65 and D39, a conservative filtered recirculation flow rate of 400 cfm was used. Sensitivity analyses of the post-LOCA dose to Control Room operators were performed at assumed unfiltered Control Room inleakage rates of 0, 3000, and 10,000 cfm (Reference 22, Items B.62, B.66, D.36, and D.40).

The EOF/TSC is automatically isolated and placed in the emergency ventilation mode upon an air intake radiation monitor isolation signal. For analysis purposes, the accident activity was allowed to enter the EOF/TSC for 2 hours into the LOCA at the normal ventilation flow rate of 4690 cfm at which time the EOF/TSC was assumed to be isolated by operator action (Reference 22, Items B.72 and B.76). Sensitivity analyses of the post-LOCA dose to the EOF/TSC were performed at assumed unfiltered EOF/TSC inleakage rates of 0, 1720, and 10,000 cfm (Reference 22, Items B.73 and B.78).

2.3.1.1.3 LOCA Removal Inputs

The accident's activity released from the core is partially removed by natural deposition mechanisms in the drywell, main steam lines, and condenser as well as by air filtration systems in the Reactor Building, Control Room, and EOF/TSC. The natural deposition removal mechanisms are characteristic of the nature of the AST and represent a change in the plant design and licensing basis.

Drywell natural deposition was simulated using the 10^{th} percentile data for the Power's natural deposition model in the RADTRAD code (Reference 7).

Main steam line pipe deposition was simulated using the RADTRAD code's (Reference 7) Brockmann – Bixler pipe deposition model using plant specific piping geometry data crediting deposition in horizontal pipe legs only.

Activity deposition in the plant condenser was estimated using a BSEP specific condenser deposition filter efficiency calculated per methodology in Reference 13.

Filter removal at the SGT system and by the Emergency modes of the Control Room and EOF/TSC Ventilation systems was simulated using plant design data as listed in Table 2-3. The SGT bypass flow assumed is the maximum allowed by plant Technical Specifications and is incorporated into the filter efficiencies shown.

An aerosol removal efficiency of 99.99% and 95% is used for the SGT and Control Room/EOF/TSC ventilation system HEPA filters, respectively (Reference 22, Items B.22, B.67, and B.80). The SGT charcoal filter efficiency of 99% for elemental and organic iodines (Reference 22, Item B.22) is consistent with Regulatory Guide 1.52 (Reference 15). The Control Room and EOF/TSC charcoal filter efficiencies of 90% for elemental iodine and 90% for organic iodine (Reference 22, Items B.67 and B.80) are consistent with the ASTM D3803-1989 requirements.

2.3.1.1.4 Personnel Dose Conversion Inputs

The standard breathing rate of 3.5E-04 m^3/sec for the 30-day accident duration (References 5 and 7) was used for onsite personnel in the Control Room and EOF/TSC dose assessments. Likewise, the standard offsite breathing rates of 3.5E-04 m^3/sec from 0 to 8-hours, 1.8E-04 m^3/sec from 8 to 24 hours, and 2.3E-04 m^3/sec from 1 to 30 days was used in dose assessments. Occupancy factors used in the Control Room and EOF/TSC were 1.0 from 0 to 1-day, 0.6 from 1 to 4 days, and 0.4 for 4 to 30 days (Reference 5).

These key inputs are summarized in Table 2-4.

2.3.1.2 MSLBA Inputs and Assumptions

The key inputs used in this analysis are included in Table 2-5.

The postulated accident assumes a double ended break (DEB) of one main steam line, with the reactor operating at LPU, outside the secondary containment with displacement of the pipe ends that permits maximum blowdown rates. The break mass released includes the line inventory plus the system mass released through the break prior to isolation. Break isolation was assumed to be 5.5 seconds based on the maximum time allowed for main steam line isolation valve closure, 5.0 seconds, and the response time for the isolation logic, 0.5 seconds (Reference 22, Item E.13). Two activity release cases corresponding to the maximum pre-accident spike and maximum equilibrium concentration allowed by plant Technical Specifications (Reference 22, Items E.8 and E.9) or 4 μCi/gm and 0.2 μCi/gm dose equivalent I-131 respectively were assumed. These released activity assumptions are consistent with the requirements of Regulatory Guide 1.183, Reference 5.

All the accident activity was assumed released instantaneously (i.e., within 5.5 seconds corresponding to the break duration) from the Turbine Building as a ground release without credit for building holdup or dilution (Reference 5).

Atmospheric dispersion coefficients were calculated based on site specific meteorology data collected between January 1996 and December 1999 (Reference 22, Item A.1.1), as documented in Calculations BNP-RAD-001 and BNP-RAD-002 (References 16 and 17).

The MSLBA analysis demonstrates that the 30-day Control Room dose is nearly insensitive to the time of Control Room isolation. Prior to isolation, the activity is assumed to enter the Control Room at the normal ventilation rate of 2100 cfm (Reference 22, Items E.22 and D.34). The Control Room 30-day MSLBA dose is reported as a bounding dose encompassing isolated and non-isolated simulations of the event. Thus, Control Room isolation need not be considered in the plant operations response to this postulated event. Sensitivity analyses of the post-MSLB dose to Control Room operators were performed at assumed unfiltered Control Room inleakage rates of 0, 3000, and 10,000 cfm (Reference 22, Items E.24, E.28, D.36, and D.40).

Filter removal by the emergency modes of the Control Room Emergency Ventilation system was simulated using plant design data as listed in Table 2-3.

2.3.1.3 Refueling Accident Inputs and Assumptions

The key inputs used in this analysis are included in Table 2-6.

This postulated Fuel Handling Accident (FHA) involves the drop of a fuel assembly on top of the reactor core during refueling operations. Based on limiting considerations, 172 fuel rods (Reference 22, Item C.5) were assumed damaged. Per Reference 22 Item C.7, a radial peaking factor of 1.50 was assumed. Consistent with plant refueling procedures, a post-shutdown 24-hour decay period (Reference 22, Item C.9) was used to determine the release activity inventory. All the gap activity in the affected rods was assumed released instantaneously into the pool. A pool water iodine decontamination factor of 200 was used, based on Reference 5, and plant configuration.

Conservatively, all the FHA activity was assumed released within two hours (Reference 5) from the Reactor Building as a ground release (Reference 22, Items C.21 and C.22) with no credit for Reactor Building holdup, Reactor Building dilution, or SGT operation.

Atmospheric dispersion coefficients were calculated based on site specific meteorology data collected between January 1996 and December 1999 (Reference 22, Item A.1.1) as documented in Calculations BNP-RAD-001 and BNP-RAD-002 (References 16 and 17).

The Control Room was conservatively assumed to be manually isolated in 20 minutes following the accident. This isolation timing is consistent with the assumptions used in the LOCA analysis. Prior to isolation, the activity was assumed to enter the control room at the normal ventilation rate of 2100 cfm (Reference 22, Item D.34). Sensitivity analyses of the post-FHA dose to Control Room operators were performed at assumed unfiltered Control Room inleakage rates of 0, 3000, and 10,000 cfm (Reference 22, Items D.36 and D.40).

Filter removal by the emergency mode of the Control Room Emergency Ventilation system was simulated using plant design data as listed in Table 2-3.

2.3.1.4 Control Rod Drop Accident Inputs and Assumptions

The key inputs used in this analysis are included in Table 2-7.

The plant design basis control rod drop accident (CRDA) involves the rapid removal of a high worth control rod resulting in a reactivity excursion that encompasses the consequences of any other postulated CRDA. Based on limiting assumptions, a total of 1200 fuel rods were assumed damaged with 0.77 % of the damaged rods clad melting (Reference 22, Items D.4 and D.5). Per Reference 22 Items D.7 and C.7, a radial peaking factor of 1.50 was assumed.

A fraction of the activity released from the damaged fuel was assumed to reach the turbine and condenser where it was assumed to be released from the Turbine Building at ground level at a rate of 1% per day for a period of 24 hours (Reference 5). No credit is taken for Turbine Building holdup or dilution (Reference 5).

Atmospheric dispersion coefficients were calculated based on site specific meteorology data collected between January 1996 and December 1999 (Reference 22, Item A.1.1) as documented in Calculations BNP-RAD-001 and BNP-RAD-002 (References 16 and 17).

The Control Room was conservatively assumed to be manually isolated in 20 minutes following the accident. This isolation timing is consistent with the assumptions used in the LOCA analysis. Prior to isolation, the activity was assumed to enter the control room at the normal ventilation rate of 2100 cfm (Reference 22, Item D.34). Sensitivity analyses of the post-CRDA dose to Control Room operators were performed at assumed unfiltered Control Room inleakage rates of 0, 3000, and 10,000 cfm (Reference 22, Items D.36 and D.40).

Filter removal by the emergency mode of the Control Room Emergency Ventilation system was simulated using plant design data as listed in Table 2-3.

2.3.2 NUREG-0737, Item II.B.2

The key inputs used in this analysis include:

- Plant-specific fission product inventories and reactor coolant source terms.
- Post-accident cloud dose as addressed in Section 3.1.1.1 of this report.
- Calculation BNP-RAD-006 (Reference 19), which evaluates post-accident limiting mission doses associated with Post-Accident Sampling System (PASS) sample collection and the SGT stack sampling based on References 11 and 16, Item F.1.

Table 2-1: Key LOCA Analysis Inputs and Assumptions, Release Inputs

Release Inputs	
Input/Assumption	**Value**
Fission Products Core Inventory	Plant-specific ORIGEN2
Fission Products Release Fractions, Form, and Timing	Per NUREG-1465 (Reference 3), Regulatory Guide 1.183 (Reference 5), and BWROG Gap Release Criteria (Reference 6)
Primary Containment Leakage Rate	0.5%/day for 24 hours, 0.4%/day afterwards (References 16 and 5)
Primary Containment Leakage, Secondary Containment Bypass (SCB)	23 scfh (Reference 22)
ESF Systems Leakage Rate	20 gpm (Reference 22)
Total MSIV Leakage	46 scfh (Reference 22)
Start of SLC system Injection	59 minutes into the LOCA (Reference 22)
SLC system Inventory	2530 gallons (Reference 22)
SLC system Injection Flow Rate	41.2 gpm (Reference 22)
SLC system Solution Composition	9 weight % Sodium Pentaborate (Reference 22)

Table 2-2: Key LOCA Analysis Inputs and Assumptions, Transport Inputs

Transport Inputs	
Input/Assumption	**Value (Reference 22)**
Reactor Building PPP	5 minutes
Reactor Building Ground PPP Release Location	Closest Point to Control Room and EOF/TSC Intake from Reactor Building South Wall
Turbine Building Ground MSIV/SCB Release Location	Low Pressure Turbine at Operating Floor
Meteorology	Site Specific, 01/96 to 12/99
Control Room Isolation	Manual at 20 minutes
Control Room Normal Mode Outside Air Intake Flow Rate	2100 cfm
Control Room Unfiltered Inleakage Rate	0, 3000, and 10,000 cfm
Technical Support Center Isolation	2 hours
EOF/TSC Normal Mode Outside Air Intake Flow Rate	4690 cfm
EOF/TSC Unfiltered Inleakage Rate	0, 1720, and 10,000 cfm

Table 2-3: Key LOCA Analysis Inputs and Assumptions, Removal Inputs

Removal Inputs	
Input/Assumption	**Value**
Drywell Natural Deposition	Power's 10^{th} Percentile Model (Reference 5)
Main Steam Lines Deposition	Brockmann - Bixler Pipe Deposition Model (Reference 5)
Condenser Deposition Elemental & Particulate Filter Efficiency	GENE Condenser Deposition Model (Reference 13 and BNP-RAD-007) 99.6%
SGT Flow Rate	2700 cfm (Reference 22)
SGT Filter Efficiency	99.99%, Aerosol 99%, Elemental and Organic (Reference 22)
Control Room Emergency Mode Outside Air Intake Flow Rate	1500 cfm (Reference 22)
Control Room Emergency Mode Filtered Recirculation Flow Rate	400 (Reference 22)
EOF/TSC Emergency Mode Outside Air Intake Flow Rate	4690 cfm (Reference 22)
Control Room & EOF/TSC Emergency Mode Filter Iodine Efficiency	95%, Aerosol (Reference 22) 90%, Elemental (Reference 22) 90%, Organic (Reference 22)

Table 2-4: Key Accident Analysis Inputs and Assumptions, Personnel Dose Conversion Inputs

Personnel Dose Conversion Inputs	
Input/Assumption	**Value**
Onsite Breathing Rate	3.5E-04 m^3/sec (References 5 and 7)
Offsite Breathing Rate	0-8 hours: 3.5E-04 m^3/sec 8-24 hours: 1.8E-04 m^3/sec 1-30 days: 2.3E-04 m^3/sec (References 5 and 7)
Control Room & EOF/TSC Occupancy Factors	0-1 day: 1.0 1-4 days: 0.6 4-30 days: 0.4 (References 5 and 7)

Table 2-5: Key MSLBA Analysis Inputs and Assumptions

MSLBA Inputs	
Input/Assumption	**Value (Reference 22)**
Mass Release Rate	Same as existing design basis
Break Isolation Time	5.5 seconds
Maximum Pre-Accident Spike Iodine Concentration	Per TS value of 4 μCi/gm
Maximum Equilibrium Iodine Concentration	Per TS value of 0.2 μCi/gm
Release Period	5.5 seconds
Turbine Building Ground Release Location	Turbine Building Exhaust Vent
Meteorology	Site Specific, 01/96 to 12/99
Control Room Isolation	See Note 1
Control Room Normal Mode Outside Air Intake Flow Rate	2100 cfm
Control Room Unfiltered Inleakage Rate	0, 3000, and 10,000 cfm

Notes:

1. The MSLBA analysis demonstrates that the 30-day Control Room dose is nearly insensitive to the time of Control Room isolation. Prior to isolation, the activity is assumed to enter the Control Room at the normal ventilation rate of 2100 cfm (Reference 22, Items E.22 and D.34). The Control Room 30-day MSLBA dose is reported as a bounding dose encompassing isolated and non-isolated simulations of the event. Thus, Control Room isolation need not be considered in the plant operation response to this postulated event.

Table 2-6: Key FHA Analysis Inputs and Assumptions

FHA Inputs	
Input/Assumption	**Value**
Number of Failed Rods	172 (Reference 22)
Radial Peaking Factor	1.50 (Reference 22)
Fuel Decay Period	24 hours (Reference 22)
Pool Water Iodine Decontamination Factor	200 (Reference 5)
Release Period	2 hours (Reference 5)
Reactor Building Ground Release Location	Reactor Building Vent (Reference 22)
Meteorology	Site Specific, 01/96 to 12/99 (Reference 22)
Manual Control Room Isolation	20 minutes (Reference 22)
Control Room Normal Mode Outside Air Intake Flow Rate	2100 cfm (Reference 22)
Control Room Unfiltered Inleakage Rate	0, 3000, and 10,000 cfm (Reference 22)

Table 2-7: Key CRDA Analysis Inputs and Assumptions

CRDA Inputs	
Input/Assumption	**Value**
Number of Failed Rods	1200 (Reference 22)
% Fuel Melt	0.77% (Reference 2)
Radial Peaking Factor	1.50 (Reference 22)
Condenser Leakage Rate	1%/day (Reference 5)
Release Period	24 hours (Reference 5)
Turbine Building Ground Release Location	Turbine Building (Reference 22)
Meteorology	Site Specific, 01/96 to 12/99 (Reference 22)
Manual Control Room Isolation	20 minutes (Reference 22)
Control Room Normal Mode Outside Air Intake Flow Rate	2100 cfm (Reference 22)
Control Room Unfiltered Inleakage Rate	0, 3000, and 10,000 cfm (Reference 22)

3. RESULTS

3.1 Evaluation Results

3.1.1 Accident Radiological Consequence Analyses

The postulated accident radiological consequence analyses were updated for both EPU and AST implementation impact.

3.1.1.1 LOCA

The radiological consequences of the design basis LOCA were analyzed using the RADTRAD code (Reference 7) and the inputs / assumptions defined in Section 2.3.1.1 of this report. The detailed analysis is documented in Calculation BNP-RAD-007 (Reference 20). The post-accident doses are the result of four distinct activity releases:

1. Primary to secondary containment leakage. This leakage is directly released into the environment during the secondary containment's positive pressure period (PPP) and filtered by the SGT afterwards.

2. Primary leakage, secondary containment bypass. This portion of the primary leakage bypasses the secondary containment and is released into the environment via the condenser.

3. ESF system leakage into the secondary containment. This leakage is also directly released into the environment during the secondary containment's PPP and filtered by the SGT afterwards.

4. MSIV leakage from the primary containment into the Main Condenser. This condenser leakage is released, undiluted and unfiltered, through the Turbine Building.

Table 3-1 presents the results of the LOCA radiological consequence analysis for offsite receptors. As indicated, the Exclusion Area Boundary (EAB), 3000 feet from the plant stack, and the Low Population Zone (LPZ), 2 miles from the site, (Reference 22, Items A.2.5 and A.2.6) calculated doses are within the regulatory limits after EPU/AST implementation.

The results of the LOCA radiological consequence analysis for the Control Room and EOF/TSC are presented in Table 3-2. The dose to both the Control Room and EOF/TSC occupants includes terms for:

1. In-leakage internal cloud immersion and inhalation contribution from the primary containment, secondary containment bypass, ESF, and MSIV leakage releases.

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

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

In addition, the total Control Room dose include a Reactor Building/Control Room shielded contribution from the SGT filters, ECCS piping, and Control Room ventilation system filters. The EOF/TSC dose includes a Reactor Building/EOF/TSC shielded contribution from the SGT filters and EOF/TSC ventilation system filters.

Per Table 3-2, the post-LOCA Control Room and EOF/TSC calculated doses are within regulatory limits after EPU/AST implementation.

The Table 3-2 post-LOCA Control Room/EOF/TSC dose corresponds to an assumed unfiltered inleakage rate of 10,000 cfm (Reference 22, Items B.62, B.66, D.36, and D.40).

3.1.1.2 MSLBA

The radiological consequences of the design basis MSLBA were analyzed in Calculation BNP-RAD-008 (Reference 21) using the RADTRAD code (Reference 7) and the inputs / assumptions defined in Section 2.3.1.2 of this report. Two activity release cases corresponding to the reactor coolant maximum pre-accident spike and maximum equilibrium concentration allowed by plant Technical Specifications of 4 μCi/gm and 0.2 μCi/gm dose equivalent I-131 respectively (Reference 22, Items E.8 and E.9) were analyzed.

Table 3-3 presents the results of the MSLBA radiological consequence analysis for offsite receptors. As indicated, both the EAB and LPZ calculated doses are within the regulatory limits after EPU/AST implementation for both cases analyzed.

The calculated Control Room doses for the MSLBA are presented in Table 3-4. The doses are bounded by the LOCA event doses and are within the regulatory limits after EPU/AST implementation for both cases analyzed.

The Table 3-4 post-MSLBA Control Room dose corresponds to an unfiltered inleakage rate of zero (0) cfm. The MSLBA Control Room unfiltered inleakage sensitivity analyses demonstrated that the post-MSLBA Control Room doses are maximized at zero inleakage rate due to the instantaneous nature of the release and the "clean-up" effect of higher inleakage rates following the release.

3.1.1.3 Refueling Accident

The radiological consequences of the design basis FHA were analyzed in Calculation BNP-RAD-008 (Reference 21) using the RADTRAD code (Reference 7) and the inputs / assumptions defined in Section 2.3.1.3 of this report. Table 3-5 presents the results of the FHA radiological consequence analysis for offsite and Control Room receptors. As indicated, both the offsite (EAB and LPZ) calculated doses and the Control Room operator doses are within the regulatory limits after EPU/AST implementation.

3.1.1.4 Control Rod Drop Accident

The radiological consequences of the design basis CRDA were analyzed in Calculation BNP-RAD-008 (Reference 21) using the RADTRAD code (Reference 7) and the inputs / assumptions defined in Section 2.3.1.4 of this report. Table 3-6 presents the results of the CRDA radiological consequence analysis for offsite and Control Room receptors. As indicated, both the offsite and Control Room calculated doses are within the regulatory limits after EPU/AST implementation.

3.1.1.5 Plant Area Post-Accident Radiation Levels

The post-accident radiation levels to plant areas, other than the personnel doses in the Control Room/EOF/TSC and NUREG-0737 post-accident mission doses addressed in this task, are relevant for EQ assessments only.

Post-accident doses for personnel in the Control Room, EOF/TSC, and NUREG-0737 mission doses were calculated based on the AST source term, as discussed in this report. However, post-accident doses for equipment qualification continue to be based on the TID-14844 source term and were scaled appropriately to account for the increased power level for EPU.

3.1.2 NUREG-0737, Item II.B.2

The post-accident limiting mission doses associated with Post Accident Sampling System (PASS) sample collection and the SGT Stack sampling have been evaluated per References 11 and 16, Item F.1. Reference 11 provides a PASS vital area access mission dose methodology that can be used for this assessment. The BSEP controlling vital area access missions were identified and their impact from the EPU and AST documented in a new calculation, BNP-RAD-006 (Reference19).

The EPU and AST impact the source terms used in the evaluation of vital area access mission doses. Specifically, Calculation BNP-RAD-006 (Reference 19) addresses the effects on mission dose evaluations from revised post-accident cloud immersion, reactor building shine, and reactor coolant / drywell atmosphere contained source doses associated with the EPU and the AST. Calculation BNP-RAD-006 (Reference 19) demonstrates that the BSEP plant shielding is adequate to maintain the post-accident vital area access considering the impact from the EPU and AST as shown in Table 3-7.

3.1.3 Atmospheric Dispersion Factors

LOCA χ/Q values are summarized below in Tables 3-8 through 3-16. Ground level and elevated release χ/Q values for the EAB and LPZ locations are taken from the PAVAN runs of Calculation BNP-RAD-001, Revision 0 (Reference 16) and are itemized within Tables 3-8 through 3-11. Ground level and elevated release χ/Q values for the Control Room and EOF/TSC locations are taken from the ARCON96 runs of Calculation BNP-RAD-002, Revision 0 (Reference 17) and are itemized within Tables 3-5 through 3-9.

For the MSLB accident, the χ/Q values for the EAB and the LPZ are the same as (see Table 3-17) for the FHA. The MSLB χ/Q's for the Control Room are taken from the ARCON96 runs of Calculation BNP-RAD-002, Revision 0 (Reference 17) and are itemized within Table 3-19; the release is ground level via the Turbine Building exhaust.

For the CRDA, the ground-level release χ/Q values for the EAB and the LPZ are the same as (see Table 3-17) for the FHA. The CRDA χ/Q's for the Control Room are taken from the ARCON96 runs of Calculation BNP-RAD-002, Revision 0 (Reference 17) and are itemized within Table 3-20; the release is ground level from the condenser. The elevated release χ/Q values for the EAB and LPZ locations are taken from the PAVAN runs of Calculation BNP-RAD-001, Revision 0 (Reference 16) and are itemized within Table 3-21. The elevated release χ/Q values for the Control Room are taken from the ARCON96 runs of Calculation BNP-RAD-002, Revision 0 (Reference 17) and are itemized within Table 3-22. Note that per Regulatory Guide 1.145, Sections 2.1.2 and 2.2.2, the BSEP site should be considered inland and, therefore, the fumigation χ/Q was used in the worst 30 minutes.

FHA χ/Q values are summarized below in Tables 3-17 and 3-18. The FHA χ/Q's for the EAB and LPZ are taken from the PAVAN runs of Calculation BNP-RAD-001, Revision 0 (Reference 16) and are itemized within Table 3-17; the release is ground level via the Reactor Building exhaust vent. The FHA χ/Q's for the Control Room are taken from the ARCON96 runs of Calculation BNP-RAD-002, Revision 0 (Reference 17) and are itemized within Table 3-18; the release is ground level via the Reactor Building exhaust vent.

3.1.4 Post-Accident Containment Water Chemistry Management

The re-evolution of elemental iodine from the suppression pool is strongly dependent on pool pH. The analysis assumed tht sodium pentaborate was injected via the SLC system within two hours of the onset of a DBA LOCA. The conservative modeling of the BSEP containment cabling produced a large amount of hydrolic acid. The minimum pool pH at 30 days post-LOCA was calculated to be 8.11, per Calculation BNP-RAD-003 (Reference 18). As this result is well above 7.0, this satisfies the conditions for minimizing the re-evolution of elemental iodine. The acid production results and pool pH, as a function of hours following the LOCA, are itemized in Table 3-23.

3.2 Evaluation Conclusions

3.2.1 Accident Radiological Consequence Analyses

As shown in Tables 3-1 through 3-6, the plant's accident radiological consequence analyses demonstrate that the post-accident offsite, Control Room, EOF/TSC doses can be maintained below regulatory limits following EPU/AST implementation. All analyses were performed at 2981 MWt (102% of 2923 MWt) and in accordance with Reference 5 for AST implementation.

3.2.2 NUREG-0737, Item II.B.2

An evaluation of the bounding post-accident vital area missions shows that the current plant shielding and post-accident sampling procedures are adequate to maintain the post-accident vital area access considering the impact from the EPU and AST.

Table 3-1: LOCA Radiological Consequence Analysis, Offsite Doses

Dose Component	EAB(1) (rem TEDE)	LPZ(2) (rem TEDE)
Primary Containment Leakage	0.24	0.13
Secondary Containment Bypass	0.08	0.36
ESF Leakage	0.13	0.12
MSIV Leakage	0.16	0.73
TOTAL	**0.61**	**1.34**
Regulatory Limit	25	25

(1) Worst 2-hour integrated dose.

(2) 30-day integrated dose.

Table 3-2: LOCA Radiological Consequence Analysis, Control Room and EOF/TSC Doses

Dose Component	Control Room(1) (rem TEDE)	EOF/TSC(1) (rem TEDE)
Primary Containment Leakage	0.22	0.04
Secondary Containment Bypass	0.51	0.08
ESF Leakage	0.25	0.08
MSIV Leakage	1.13	0.17
External Cloud	0.01	0.04
Reactor Building Direct Shine	0.36	0.36
SGT Filter Direct Shine	0.18	0.18
Control Room/EOF/TSC Filter Shine	0.64	0.09
ECCS Piping Shine	0.10	Negligible
TOTAL	**3.40**	**1.04**
Regulatory Limit	5	5

(1) Assumes conservative unfiltered inleakage of 10,000 cfm (Reference 16).

Table 3-3: MSLBA Radiological Consequence Analysis, Offsite Doses

Case	EAB[1] (rem TEDE)	LPZ[2] (rem TEDE)	Regulatory Limit (rem TEDE)
4 μCi/gm dose equivalent I-131	2.52	0.89	25
0.2 μCi/gm dose equivalent I-131	1.27E-01	4.50E-02	2.5

(1) Worst 2-hour integrated dose.

(2) 30-day integrated dose.

Table 3-4: MSLBA Radiological Consequence Analysis, Control Room Doses

Case	Control Room Dose[1] (rem TEDE)	Regulatory Limit (rem TEDE)
4 μCi/gm dose equivalent I-131	0.5	5
0.2 μCi/gm dose equivalent I-131	2.50E-02	5

(1) Assumes a conservative unfiltered inleakage of 0 cfm.

Table 3-5: FHA Radiological Consequence Analysis, Offsite and Control Room Doses

Event	EAB(1) (rem TEDE)	LPZ(2) (rem TEDE)	Control Room Dose(3) (rem TEDE)
Refueling Accident Inside Containment	5.51	1.95	2.69
Regulatory Limit	6.25	6.25	5

(1) Worst 2-hour integrated dose.

(2) 30-day integrated dose.

(3) Assumes a conservative unfiltered inleakage of 10,000 cfm.

Table 3-6: CRDA Radiological Consequence Analysis, Offsite and Control Room Doses

Event	EAB$^{(1)}$ (rem TEDE)	LPZ$^{(2)}$ (rem TEDE)	Control Room Dose$^{(3)}$ (rem TEDE)
Control Rod Drop Accident	0.27	0.22	0.28
Regulatory Limit	6.25	6.25	5

(1) Worst 2-hour integrated dose.

(2) 30-day integrated dose.

(3) Assumes a conservative unfiltered inleakage of 10,000 cfm.

Table 3-7: NUREG-0737, Item II.B.2 Assessment

Parameter	Whole Body Dose(1) (rem TEDE)
PASS Sampling Mission Dose	4.45
SGT Stack Sampling Mission Dose	4.71
Regulatory Limit	5

(1) Total mission integrated dose.

Table 3-8: Maximum Offsite Ground Level χ/Q for EAB Distance of 3000 Feet

Time Period	χ/Q (sec/m^3)
0-2 hours	2.20E-03
0-8 hours	1.23E-03
8-24 hours	9.26E-04
1-4 days	4.96E-04
4-30 days	2.02E-04

Table 3-9: Maximum Offsite Ground Level χ/Q for LPZ Distance of 2 Miles

Time Period	χ/Q (sec/m^3)
0-2 hours	7.77E-04
0-8 hours	3.36E-04
8-24 hours	2.21E-04
1-4 days	8.90E-05
4-30 days	2.41E-05

Table 3-10: Maximum Offsite Elevated χ/Q for EAB Distance of 3000 Feet

Time Period	χ/Q (sec/m^3)
Fumigation	5.85E-05
0-2 hours	3.63E-06
0-8 hours	2.04E-06
8-24 hours	1.53E-06
1-4 days	9.27E-07
4-30 days	4.80E-07

Table 3-11: Maximum Offsite Elevated χ/Q for LPZ Distance of 2 Miles

Time Period	χ/Q (sec/m^3)
Fumigation	1.88E-05
0-2 hours	3.50E-06
0-8 hours	1.92E-06
8-24 hours	1.42E-06
1-4 days	7.40E-07
4-30 days	3.07E-07

Table 3-12: Ground Level Release – Reactor Building Wall to Control Room χ/Q's

Time Period	χ/Q (sec/m^3)
0-2 hours	4.05E-03
2-8 hours	3.67E-03
8-24 hours	1.74E-03
1-4 days	1.44E-03
4-30 days	1.02E-03

Table 3-13: Ground Level Release – Reactor Building Wall to EOF/TSC χ/Q's

Time Period	χ/Q (sec/m^3)
0-2 hours	3.11E-04
2-8 hours	2.55E-04
8-24 hours	1.15E-04
1-4 days	8.57E-05
4-30 days	5.52E-05

Table 3-14: Ground Level Release – Condenser to Control Room χ/Q's

Time Period	χ/Q (sec/m^3)
0-2 hours	1.15E-03
2-8 hours	1.03E-03
8-24 hours	4.90E-04
1-4 days	4.11E-04
4-30 days	2.63E-04

Table 3-15: Ground Level Release – Condenser to EOF/TSC χ/Q's

Time Period	χ/Q (sec/m^3)
0-2 hours	2.04E-04
2-8 hours	1.90E-04
8-24 hours	9.89E-05
1-4 days	6.96E-05
4-30 days	4.88E-05

Table 3-16: Control Room and EOF/TSC Elevated Release χ/Q's

Time Period	Control Room χ/Q (sec/m^3)	EOF/TSC χ/Q (sec/m^3)
Fumigation	3.19E-04	2.16E-04
0-2 hours	4.65E-06	3.74E-06
0-8 hours	1.73E-06	1.25E-06
8-24 hours	1.06E-06	7.23E-07
1-4 days	3.62E-07	2.21E-07
4-30 days	7.76E-08	4.01E-08

Table 3-17: Offsite Ground Level Release χ/Q's

Time Period	EAB χ/Q (sec/m^3)	LPZ χ/Q (sec/m^3)
0 – 2 hours	2.20E-03	7.77E-04
2 - 8 hours	1.23E-03	3.36E-04
8 - 24 hours	9.26E-04	2.21E-04
1 - 4 days	4.96E-04	8.90E-05
4 - 30 days	2.02E-04	2.41E-05

Table 3-18: Ground Level Release χ/Q's From Reactor Building Vent to Control Room

Time Period	Control Room χ/Q (sec/m^3)
0 – 2 hours	1.48E-03
2 - 8 hours	1.30E-03
8 - 24 hours	6.74E-04
1 - 4 days	4.90E-04
4 - 30 days	3.49E-04

Table 3-19: Ground Level Release χ/Q's from the Turbine Building Exhaust

Time Period	Bounding χ/Q – Turbine Building Exhaust to Control Room Intake (sec/m^3)
0 – 2 hours	5.55E-04
2 – 8 hours	4.41E-04
8 – 24 hours	2.21E-04
1 - 4 days	1.54E-04
4 – 30 days	1.21E-04

Table 3-20: Ground Level Release χ/Q's From Condenser to Control Room

Time Period	Control Room χ/Q (sec/m^3)
0 - 2 hours	1.15E-03
2 - 8 hours	1.03E-03
8 - 24 hours	4.90E-04
1 - 4 days	4.11E-04
4 - 30 days	2.63E-04

Table 3-21: Elevated Release χ/Q's From Stack to EAB and LPZ

Time Period	EAB χ/Q (sec/m^3)	LPZ χ/Q (sec/m^3)
Fumigation	5.85E-05	1.88E-05
0 – 2 hours	3.63E-06	3.50E-06
0 - 8 hours	2.04E-06	1.92E-06
8 - 24 hours	1.53E-06	1.42E-06
1 - 4 days	8.26E-07	7.40E-07
4 - 30 days	3.40E-07	2.90E-07

Table 3-22: Elevated χ/Q's From Stack to Control Room

Time Period	Control Room χ/Q (sec/m^3)
Fumigation	3.19E-04
0 – 2 hours	4.65E-06
0 - 8 hours	1.73E-06
8 - 24 hours	1.06E-06
1 - 4 days	3.62E-07
4 - 30 days	7.76E-08

Table 3-23: Acid Generation and Pool pH Results

Time (Hours)	HI (moles)	HNO_3 (moles)	HCl (moles)	pH
2	1.01	3.35	5.40	8.63
4	1.02	9.32	179	8.60
8	1.02	16.3	320	8.57
16	1.02	26.7	530	8.53
24	1.02	35.1	691	8.50
48	1.02	54.6	1,034	8.43
72	1.02	70.0	1,267	8.38
120	1.02	94.9	1,580	8.32
168	1.02	115	1,788	8.28
240	1.02	141	2,001	8.23
480	1.02	204	2,370	8.15
720	1.02	249	2,548	8.11

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16. Calculation BNP-RAD-001, "Accident Offsite Radiological Atmospheric Dispersion Factors (Chi/Q)," Rev. 0.

17. Calculation BNP-RAD-002, "Accident Control Room and TSC Radiological Atmospheric Dispersion Factors (Chi over Q)," Rev. 0.

18. Calculation BNP-RAD-003, "Suppression Pool Post-LOCA pH Calculation With Alternate Source Term," Rev. 0.

19. Calculation BNP-RAD-006, "NUREG-0737 Item II.B.2 – Mission Dose Assessment for AEP and AST," Rev. 0.

20. Calculation BNP-RAD-007, "DBA LOCA Radiological Dose with Alternate Source Term," Rev. 0.

21. Calculation BNP-RAD-008, "Non-LOCA Radiological Consequence Dose with Alternate Source Term," Rev. 0.

22. Calculation BNP-RAD-010, "Design Inputs for Accident Radiological Analyses," Rev. 0.

ENCLOSURE 3

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ALTERNATIVE RADIOLOGICAL SOURCE TERM

10 CFR 50.92 Evaluation

Carolina Power & Light (CP&L) Company is requesting a revision to the Technical Specifications (TS) for Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. These changes support a full-scope application of an Alternative Source Term for BSEP, Units 1 and 2. The Alternative Source Term analyses were performed following the guidance in Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors," dated July 2000, and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms." In order to support a planned Extended Power Uprate of BSEP, Units 1 and 2, the new Alternative Source Term analyses have been performed at 102 percent of the uprated power level (i.e., 2981 megawatts thermal (MWt)).

The Alternative Source Term analyses have been performed without crediting secondary containment operability during fuel handling accidents. As such, the proposed license amendments relax operability requirements, during fuel handling and core alterations, for: (1) the Secondary Containment system, (2) Secondary Containment Isolation Instrumentation, (3) Secondary Containment Isolation Dampers, (4) the Standby Gas Treatment system, and (5) Control Room Emergency Ventilation system isolation instrumentation. These changes are consistent with Technical Specification Task Force (TSTF) 51, Revision 2, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," which was approved by the NRC on November 1, 1999.

CP&L has concluded that the proposed changes to the TS for BSEP, Units 1 and 2, do not involve a Significant Hazards Consideration. In support of the No Significant Hazards determination, an evaluation of each of the three (3) standards set forth in 10 CFR 50.92 is provided below.

1. The proposed license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The BSEP systems affected by implementation of the Alternative Source Term analyses and the relaxations associated with TSTF-51, Revision 2, are not initiators of any design basis accidents. Therefore, because design bases accident initiators are not being altered by adoption of the Alternative Source Term analyses and the relaxations associated with TSTF-51, Revision 2, the probability of an accident previously evaluated is not affected.

The Alternative Source Term does not affect the design or normal operation of the facility. Rather, once the occurrence of the accident has been postulated, the Alternative Source Term is an input used to evaluate the consequences of an accident. Implementation of the Alternative Source Term has been evaluated for the limiting design basis accidents at BSEP, and it has been demonstrated that the dose consequences of those limiting design bases accidents are within the regulatory guidance provided by the NRC in Regulatory Guide 1.183 and Standard Review Plan Section 15.0.1. For a fuel handling accident, the AST analyses demonstrate acceptable doses, within regulatory limits, without credit for secondary containment or automatic isolation of the Control Room. As such, the consequences of an accident previously evaluated are not affected.

Based on the above, the proposed license amendments do not involve an increase in the probability or consequences of an accident previously evaluated.

2. The proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The BSEP systems affected by implementing the Alternative Source Term changes and the changes associated with TSTF-51, Revision 2, do not alter any design bases accident initiators or create new types of accident precursors. In addition, these changes do not affect the design function or mode of operation of systems, structures, or components in the facility such that new equipment failure modes are created. Therefore, the proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendments do not involve a significant reduction in a margin of safety.

   The changes proposed are associated with the implementation of a new licensing basis for BSEP. Approval of the change from the original source term, developed in accordance with TID-14844, to a new Alternative Source Term, as described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, Final Report," dated February 1, 1995, is being requested. The results of the accident analyses, revised in support of the proposed license amendments, are subject to revised acceptance criteria. These analyses have been performed using conservative methodologies, as specified in Regulatory Guide 1.183. Safety margins have been evaluated and margin has been retained to ensure that the analyses adequately bound the postulated limiting event scenarios. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67, "Alternative source term," and Regulatory Guide 1.183.

   The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries, as well as the Control Room and Emergency Operations Facility/Technical Support Center, are within corresponding regulatory limits. Specifically, the margin of safety for these accidents is considered to be that provided by meeting the applicable regulatory limits, which for three of five event scenarios (i.e., the control rod drop accident, the fuel handling accident, and one of the two limits for a main steam line break accident), is conservatively set below the 10 CFR 50.67 limit. With respect to the Control Room personnel doses, the margin of safety is the difference between the 10 CFR 50.67 limits and the regulatory limit defined by 10 CFR 50, Appendix A, General Design Criterion 19.

Since the proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries, as well as the Control Room are within corresponding regulatory limits, the proposed license amendments do not involve a significant reduction in a margin of safety.

ENCLOSURE 4

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ALTERNATIVE RADIOLOGICAL SOURCE TERM

Environmental Considerations

Identification of the Proposed Action

Carolina Power & Light (CP&L) Company is requesting a revision to the Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. These changes support a full-scope application of an Alternative Source Term (AST) for BSEP, Units 1 and 2.

The license amendment application revises the BSEP, Unit 1 and 2 Technical Specifications to implement various assumptions in the Alternative Source Term analyses. The proposed license amendments provide for a change in design and licensing bases for full-scope application of the AST in accordance with 10 CFR 50.67, "Accident source term,"' to evaluations of the consequences of a design-basis loss-of-coolant accident (LOCA), main steam line break accident, control rod drop accident, and fuel handling accident. The proposed license amendments also revise the BSEP, Unit 1 and 2 Technical Specifications by relaxing operability requirements for secondary containment, secondary containment isolation instrumentation, secondary containment isolation dampers, the Standby Gas Treatment (SGT) system, and isolation instrumentation for the Control Room Emergency Ventilation (CREV) system during core alterations or movement of irradiated fuel assemblies. The license amendment application describes the results of radiological analysis of the four limiting design-basis accidents for BSEP, Units 1 and 2.

The evaluation of the radiological consequences for the identified design basis accidents apply the AST consistent with the guidance in NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors," dated July 2000, and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms."

Need For The Proposed Action

The proposed revisions to change the licensing and design basis to implement the AST analyses are being pursued to support Extended Power Uprate of BSEP, Units 1 and 2. In order to support the Extended Power Uprate, the new AST analyses have been performed at 102 percent of a maximum power level of 2923 megawatts thermal (i.e., 2981 MWt). In addition, the proposed revisions relax operability requirements for secondary containment, secondary containment isolation instrumentation, secondary containment isolation dampers, the SGT system, and isolation instrumentation for the CREV system when core alterations or movement of irradiated fuel assemblies are occurring, thus providing increased flexibility in scheduling and conducting refueling activities. CP&L has assessed the environmental impacts associated with the proposed license amendment application and concluded that there are no significant environmental impacts associated with the proposed license amendment. The basis for CP&L's determination is summarized below.

Environmental Impacts of the Proposed Action

In December 1999, the NRC issued 10 CFR 50.67, which provides a mechanism for licensees of power reactors to replace the traditional radiological source term used in the design-basis accident analyses with an AST. The NRC also issued, in July 2000, Regulatory Guide 1.183 to provide guidance for implementing these ASTs. Section 50.67 provides that a licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall submit an application for a license amendment containing an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report. In each of these analyses for BSEP, Units 1 and 2, new atmospheric dispersion values have been derived from additional meteorology data taken over the period from January 1996 to December 1999. In addition, while existing Technical Specification requirements for the automatic isolation of the Control Room following a LOCA, main steam line break, or control rod drop accident are being retained, the AST analyses have been performed assuming manual initiation of Control Room isolation at 20 minutes following a LOCA event, main steam line break, or control rod drop accident.

As previously stated, CP&L has evaluated implementation of ASTs with respect to four design basis radiological consequence analyses previously analyzed in the BSEP Updated Final Safety Analysis Report.

Loss-Of-Coolant Accident:

The LOCA analysis postulates an instantaneous severance of a reactor recirculation system pipe. This assumption results in the most rapid coolant loss and depressurization, with coolant being discharged from both ends of the severed pipe. The release inventory has been developed assuming a maximum power level of 2981 MWt (i.e., 102 percent of the uprated power level) rather than the current maximum power level of 2558 MWt. The accident release activity has been developed using the NRC recommended guidance in Regulatory Guide 1.183 and NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, Final Report." The accident release activity has been assumed to leak from the primary containment to the Reactor Building at a rate of 0.5 percent per day. This leak rate has been reduced to 0.4 percent per day after 24 hours, as allowed by NRC Regulatory Guide 1.183. A portion of this leakage has been assumed to bypass the Reactor Building and be released to the environment. Emergency Safety Feature system valves have been conservatively assumed to leak at a rate of 20 gallons per minute, which is 20 times higher than currently established procedural action limits. The main steam line isolation valves have been assumed to leak to the environment at the maximum rate allowed by the plant's Technical Specifications for the duration of the postulated accident. CP&L has credited the deposition of radioactive iodine in the suppression pool, as allowed by Regulatory Guide 1.183, based on the pH of the suppression pool being maintained above 7. Post-accident operation of the Standby Liquid Control system has been shown to maintain the suppression pool pH above 7 for the postulated 30-day duration of the accident.

Main Steam Line Break Accident:

This design basis main steam line break accident postulates the complete severance of one main steam line outside of the secondary containment (i.e., the Reactor Building). This failure of a main steam line outside the drywell and Reactor Building represents a potential direct escape route from the reactor core to the site environs without passage through the primary containment or the Reactor Building. The steam line break accident is assumed to occur at the time the off-gas release rate is at a maximum. Coincident with this is the maximum primary coolant iodine activity of 4 μCi/ml (i.e., maximum pre-accident spike) and 0.2 μCi/ml (i.e., maximum equilibrium concentration). The release inventory has been developed assuming 102 percent of the uprated power level (i.e., 2981 MWt) rather than the current maximum power level of 2558 MWt. All accident inventory is assumed to be released until the break is terminated by closure of the main steam isolation valves (i.e., within 5.5 seconds after the break).

Control Rod Drop Accident:

The control rod drop accident postulates the rapid removal of a control rod resulting in a reactivity excursion. A total of 1200 fuel rods are assumed to be damaged with 0.77 percent of the damaged fuel rods experiencing cladding melt. A portion of the activity released from the damaged fuel is assumed to reach the main condenser, where it is released to the environment. The new analysis does not credit holdup or dilution of the activity release by the Turbine Building.

Fuel Handling Accident:

The fuel handling accident analysis postulates that a spent fuel assembly is dropped from 30 feet above the top of the reactor core during refueling operations, resulting in the breaching of the cladding for 172 fuel rods. The drop over the reactor core is more limiting (i.e., damages more fuel rods) than any drops that could occur over the fuel pool. Consistent with BSEP refueling procedures, a post-shutdown period of 24 hours is credited for radioactive decay in determining the release activity inventory. All the activity in the gap between the fuel pellets and the cladding of the damaged fuel rods is assumed to be released instantaneously into the pool. A pool water iodine decontamination factor of 200 is used, which is higher than the value of 100 used in the existing licensing basis analysis. CP&L has assumed no decontamination for noble gases released in the pool and full retention of all aerosol and particulate fission products by the pool water. Any activity leaving the fuel pool enters the Reactor Building. All of the fuel handling accident activity is assumed to be released within 2 hours from the Reactor Building as a ground release, with no credit for holdup or dilution by the Reactor Building, and no credit for operation of the SGT system. Not crediting any dilution, holdup, or cleanup by the SGT system of the activity released from the pool represents a more conservative basis than that used in the existing licensing basis fuel handling accident analysis. In addition, for analysis purposes, the Control Room has been assumed to be manually isolated at 20 minutes following the beginning of the accident rather than automatically isolating when a Control Room air intake radiation monitor isolation signal is received. This results in conservative dose assumptions for Control Room personnel.

Assessment of Offsite Consequences:

As part of the implementation of the 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, "Reactor Site Criteria - Determination of Exclusion Area, Low Population Zone, and Population Center Distance," and General Design Criterion (GDC) 19 of 10 CFR Part 50, Appendix A.

For the LOCA, the results of these AST analyses indicate that the dose at the exclusion area boundary would be no more than 0.61 rem TEDE and the dose at the low-population zone would be no more than 1.34 rem TEDE. These results are less than the TEDE criterion of 25 rem set forth in Regulatory Guide 1.183, Table 6 and, therefore, are acceptable.

For the main steam line break accident with equilibrium reactor coolant iodine activity, the results of these AST analyses indicate that the dose at the exclusion area boundary would be no more than 0.127 rem TEDE and the dose at the low-population zone would be no more than 0.045 rem TEDE. These results are less than the TEDE criterion of 2.5 rem set forth in Regulatory Guide 1.183, Table 6 and, therefore, are acceptable. For the main steam line break accident with a pre-accident reactor coolant iodine activity spike, the results of these AST analyses indicate that the dose at the exclusion area boundary would be no more than 2.52 rem TEDE and the dose at the low-population zone would be no more than 0.89 rem TEDE. These results are less than the TEDE criterion of 25 rem set forth in Regulatory Guide 1.183, Table 6 and, therefore, are acceptable.

For the control rod drop accident, the results of these AST analyses indicate that the dose at the exclusion area boundary would be no more than 0.27 rem TEDE and the dose at the low-population zone would be no more than 0.22 rem TEDE. These results are less than the TEDE criterion of 6.25 rem set forth in Regulatory Guide 1.183, Table 6 and, therefore, are acceptable.

For the fuel handling accident, the results of these AST analyses indicate that the dose at the exclusion area boundary would be no more than 5.51 rem TEDE and the dose at the low-population zone would be no more than 1.95 rem TEDE. These results are less than the TEDE criterion of 6.25 rem set forth in Regulatory Guide 1.183, Table 6 and, therefore, are acceptable.

On this basis, the proposed license amendment application to change the Technical Specifications and the licensing and design bases regarding the design-basis fuel handling accident does not represent a significant offsite radiological impact to the human environment.

Assessment of Onsite Consequences:

Using the AST and the updated atmospheric dispersion values, CP&L has evaluated the dose to operators in the Control Room for the LOCA, control rod drop accident, and fuel handling accident assuming that operators manually actuate Control Room isolation within 20 minutes. CP&L plans to revise plant procedures to enforce the AST analysis assumption of manual initiation of Control Room isolation at 20 minutes following the LOCA or control rod drop accident. For the fuel handling accident, the AST analyses demonstrate that a 24-hour decay period is sufficient to ensure secondary containment and Control Room automatic isolation are not required during core alterations or fuel handling. Plant procedures are being revised to allow manual, instead of automatic, isolation of the control room following a fuel handling accident. For the main steam line break event, the calculated Control Room dose bounds cases with a range of Control Room isolation times as well as non-isolation of the Control Room. CP&L has also evaluated these accident doses to personnel in the onsite emergency response facilities (i.e., the combined Emergency Operations Facility (EOF) and Technical Support Center (TSC)).

For the LOCA, the results indicate that the Control Room operators would receive no more than 3.4 rem TEDE and EOF/TSC personnel would receive no more than 1.04 rem TEDE. These doses are less than the TEDE limit of 5 rem contained in 10 CFR 50.67.

For the main steam line break accident with equilibrium reactor coolant iodine activity, the results indicate that the Control Room operators would receive no more than 0.025 rem TEDE. This dose is less than the TEDE limit of 5 rem contained in 10 CFR 50.67. For the main steam line break accident with a pre-accident reactor coolant iodine activity spike, the results indicate that the Control Room operators would receive no more than 0.5 rem TEDE. These doses are less than the TEDE limit of 5 rem contained in 10 CFR 50.67.

For the control rod drop accident, the results indicate that the Control Room operators would receive no more than 0.28 rem TEDE and EOF/TSC personnel would receive no more than 0.79 rem TEDE. This dose is less than the TEDE limit of 5 rem contained in 10 CFR 50.67.

For the fuel handling accident, the results indicate that the Control Room operators would receive no more than 2.69 rem TEDE. This dose is less than the TEDE limit of 5 rem contained in 10 CFR 50.67.

On these bases, the proposed license amendment application will not result in a significant onsite radiological impact to the human environment.

Conclusion:

Based on the information described above, CP&L has concluded that there are no significant environmental impacts associated with the proposed license amendment application.

ENCLOSURE 5

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ALTERNATIVE RADIOLOGICAL SOURCE TERM

Page Change Instructions

UNIT 1	
Removed page	Inserted page
1.1-2	1.1-2
1.1-3	1.1-3
3.3-62	3.3-62
3.3-63	3.3-63
3.6-29	3.6-29
3.6-30	3.6-30
3.6-31	3.6-31
3.6-33	3.6-33
3.6-34	3.6-34
3.6-35	3.6-35
3.6-36	3.6.36

UNIT 2	
Removed page	Inserted page
1.1-2	1.1-2
1.1-3	1.1-3
3.3-62	3.3-62
3.3-63	3.3-63
3.6-29	3.6-29
3.6-30	3.6-30
3.6-31	3.6-31
3.6-33	3.6-33
3.6-34	3.6-34
3.6-35	3.6-35
3.6-36	3.6-36

ENCLOSURE 6

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ALTERNATIVE RADIOLOGICAL SOURCE TERM

Typed Technical Specification Pages - Unit 1

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
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. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. 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 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 dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation,

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	Submersion, and Ingestion," 1989 and FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ISOLATION INSTRUMENTATION RESPONSE TIME	The ISOLATION INSTRUMENTATION RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves receive the isolation signal (e.g., de-energization of the MSIV solenoids). The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

(continued)

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level —Low Level 2	1,2,3,	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 101 inches
2. Drywell Pressure —High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 1.8 psig
3. Reactor Building Exhaust Radiation —High	1,2,3, (a),(b)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 16 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During movement of recently irradiated fuel assemblies in secondary containment.

3.3 INSTRUMENTATION

3.3.7.1 Control Room Emergency Ventilation (CREV) System Instrumentation

LCO 3.3.7.1 Two channels per trip system of the Control Building Air Intake Radiation—High Function shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Place one CREV subsystem in the radiation/smoke protection mode of operation.	7 days
B. CREV System initiation capability not maintained.	B.1 Place one CREV subsystem in the radiation/smoke protection mode of operation.	1 hour

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours
C. Secondary containment inoperable during movement of recently irradiated fuel assemblies in the secondary containment, or during OPDRVs.	C.1 --------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment. <u>AND</u>	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	24 months
SR 3.6.4.1.2	Verify one secondary containment access door is closed in each access opening.	24 months
SR 3.6.4.1.3	Verify each SGT subsystem can maintain ≥ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate ≤ 3000 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Dampers (SCIDs)

LCO 3.6.4.2 Each SCID shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

------------------------------------NOTES------------------------------------

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIDs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more penetration flow paths with one SCID inoperable.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic damper, closed manual damper, or blind flange.	8 hours
	<u>AND</u>		
			(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	D.1	--------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND		
	D.2	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.2.1	Verify the isolation time of each automatic SCID is within limits.	24 months
SR 3.6.4.2.2	Verify each automatic SCID actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable in MODE 1, 2 or 3.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met. OR Two SGT subsystems inoperable in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One SGT subsystem inoperable during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	C.1	Restore SGT subsystem to OPERABLE status.	31 days
D. Required Action and associated Completion Time of Condition C not met.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------		
	D.1	Place OPERABLE SGT subsystem in operation.	Immediately
	OR		
	D.2.1	Suspend movement of recently irradiated fuel assemblies in secondary containment.	Immediately
	AND		
	D.2.2	Initiate action to suspend OPDRVs.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two SGT subsystems inoperable during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	E.1 --------NOTE-------- LCO 3.0.3 is not applicable. -------------------- Suspend movement of recently irradiated fuel assemblies in secondary containment.	Immediately
	AND	
	E.2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months

ENCLOSURE 7

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ALTERNATIVE RADIOLOGICAL SOURCE TERM

<u>Typed Technical Specification Pages - Unit 2</u>

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
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. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. 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 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 dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation,

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	Submersion, and Ingestion," 1989 and FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ISOLATION INSTRUMENTATION RESPONSE TIME	The ISOLATION INSTRUMENTATION RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves receive the isolation signal (e.g., de-energization of the MSIV solenoids). The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

(continued)

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level —Low Level 2	1,2,3,	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 101 inches
2. Drywell Pressure —High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 1.8 psig
3. Reactor Building Exhaust Radiation —High	1,2,3, (a),(b)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 16 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During movement of recently irradiated fuel assemblies in secondary containment.

3.3 INSTRUMENTATION

3.3.7.1 Control Room Emergency Ventilation (CREV) System Instrumentation

LCO 3.3.7.1 Two channels per trip system of the Control Building Air Intake Radiation—High Function shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Place one CREV subsystem in the radiation/smoke protection mode of operation.	7 days
B. CREV System initiation capability not maintained.	B.1 Place one CREV subsystem in the radiation/smoke protection mode of operation.	1 hour

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Secondary containment inoperable during movement of recently irradiated fuel assemblies in the secondary containment, or during OPDRVs.	C.1 --------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment. AND	 Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	24 months
SR 3.6.4.1.2	Verify one secondary containment access door is closed in each access opening.	24 months
SR 3.6.4.1.3	Verify each SGT subsystem can maintain $\geq$ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate $\leq$ 3000 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Dampers (SCIDs)

LCO 3.6.4.2 Each SCID shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

---NOTES---

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIDs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more penetration flow paths with one SCID inoperable.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic damper, closed manual damper, or blind flange.	8 hours
	<u>AND</u>		
			(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	D.1 --------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	D.2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.2.1	Verify the isolation time of each automatic SCID is within limits.	24 months
SR 3.6.4.2.2	Verify each automatic SCID actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable in MODE 1, 2 or 3.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met. OR Two SGT subsystems inoperable in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One SGT subsystem inoperable during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	C.1 Restore SGT subsystem to OPERABLE status.	31 days
D. Required Action and associated Completion Time of Condition C not met.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	D.1 Place OPERABLE SGT subsystem in operation.	Immediately
	OR	
	D.2.1 Suspend movement of recently irradiated fuel assemblies in secondary containment.	Immediately
	AND	
	D.2.2 Initiate action to suspend OPDRVs.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two SGT subsystems inoperable during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	E.1 --------NOTE-------- LCO 3.0.3 is not applicable. -------------------- Suspend movement of recently irradiated fuel assemblies in secondary containment.	Immediately
	AND	
	E.2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months

ENCLOSURE 8

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ALTERNATIVE RADIOLOGICAL SOURCE TERM

Marked-Up Technical Specification Pages - Unit 1

Inserts Associated With Technical Specification Markups

The following insert is to be used in association with the attached Unit 1 Technical Specification markups.

<u>Insert A</u> (Page 1.1-2)

The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989 and FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

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. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

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 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 following thyroid dose conversion factors used for this calculation are defined equivalent to 1 microcurie of I-131 as~~

Insert A

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	~~determined from Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites":~~ ~~I-132, 28 microcuries;~~ ~~I-133, 3.7 microcuries;~~ ~~I-134, 59 microcuries; and~~ ~~I-135, 12 microcuries.~~
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ISOLATION INSTRUMENTATION RESPONSE TIME	The ISOLATION INSTRUMENTATION RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves receive the isolation signal (e.g., de-energization of the MSIV solenoids). The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

(continued)

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level —Low Level 2	1,2,3,	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 101 inches
2. Drywell Pressure —High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 1.8 psig
3. Reactor Building Exhaust Radiation —High	1,2,3, (a),(b)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 16 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During ~~CORE ALTERATIONS and during~~ movement of recently irradiated fuel assemblies in secondary containment.

3.3 INSTRUMENTATION

3.3.7.1 Control Room Emergency Ventilation (CREV) System Instrumentation

LCO 3.3.7.1 Two channels per trip system of the Control Building Air Intake Radiation—High Function shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Place one CREV subsystem in the radiation/smoke protection mode of operation.	7 days
B. CREV System initiation capability not maintained.	B.1 Place one CREV subsystem in the radiation/smoke protection mode of operation.	1 hour

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Secondary containment inoperable during movement of recently irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	C.1 --------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment. AND	 Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	~~C.2 Suspend CORE ALTERATIONS.~~ ~~AND~~ C.~~3~~ 2 Initiate action to suspend OPDRVs.	~~Immediately~~ Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	24 months
SR 3.6.4.1.2	Verify one secondary containment access door is closed in each access opening.	24 months
SR 3.6.4.1.3	Verify each SGT subsystem can maintain $\geq$ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate $\leq$ 3000 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Dampers (SCIDs)

LCO 3.6.4.2 Each SCID shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

-----NOTES-----

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIDs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more penetration flow paths with one SCID inoperable.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic damper, closed manual damper, or blind flange.	8 hours
	<u>AND</u>		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of recently irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	D.1 --------NOTE--------- LCO 3.0.3 is not applicable. ------------------ Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	~~AND~~	
	~~D.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND	
	D.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.2.1	Verify the isolation time of each automatic SCID is within limits.	24 months
SR 3.6.4.2.2	Verify each automatic SCID actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

recently

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable in MODE 1, 2 or 3.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met. OR Two SGT subsystems inoperable in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One SGT subsystem inoperable during movement of recently irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	C.1 Restore SGT subsystem to OPERABLE status.	31 days
D. Required Action and associated Completion Time of Condition C not met.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	D.1 Place OPERABLE SGT subsystem in operation.	Immediately
	OR	
	D.2.1 Suspend movement of recently irradiated fuel assemblies in secondary containment.	Immediately
	AND	
	~~D.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	D.2.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two SGT subsystems inoperable during movement of recently irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	E.1 --------NOTE-------- LCO 3.0.3 is not applicable. ------------------------ Suspend movement of recently irradiated fuel assemblies in secondary containment.	Immediately
	AND	
	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	E.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months

ENCLOSURE 9

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ALTERNATIVE RADIOLOGICAL SOURCE TERM

Marked-Up Technical Specification Pages - Unit 2

Inserts Associated With Technical Specification Markups

The following insert is to be used in association with the attached Unit 2 Technical Specification markups.

Insert A (Page 1.1-2)

The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989 and FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
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. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. 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 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 INSERT A	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 following thyroid dose conversion factors used for this calculation are defined equivalent to 1 microcurie of I 131 as~~

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	~~determined from Table III of TID 14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites":~~ ~~I-132, 28 microcuries;~~ ~~I-133, 3.7 microcuries;~~ ~~I-134, 59 microcuries; and~~ ~~I-135, 12 microcuries.~~
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ISOLATION INSTRUMENTATION RESPONSE TIME	The ISOLATION INSTRUMENTATION RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves receive the isolation signal (e.g., de-energization of the MSIV solenoids). The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be: a. Identified LEAKAGE 1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

(continued)

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level —Low Level 2	1,2,3,	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 101 inches
2. Drywell Pressure —High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 1.8 psig
3. Reactor Building Exhaust Radiation —High	1,2,3, (a),(b)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 16 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During ~~CORE ALTERATIONS and during~~ movement of recently irradiated fuel assemblies in secondary containment.

3.3 INSTRUMENTATION

3.3.7.1 Control Room Emergency Ventilation (CREV) System Instrumentation

LCO 3.3.7.1 Two channels per trip system of the Control Building Air Intake Radiation—High Function shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Place one CREV subsystem in the radiation/smoke protection mode of operation.	7 days
B. CREV System initiation capability not maintained.	B.1 Place one CREV subsystem in the radiation/smoke protection mode of operation.	1 hour

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1	Restore secondary containment to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours
C. Secondary containment inoperable during movement of recently irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	C.1	--------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	~~C.2 Suspend CORE ALTERATIONS.~~ ~~AND~~ C.~~3~~ 2 Initiate action to suspend OPDRVs.	~~Immediately~~ Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	24 months
SR 3.6.4.1.2	Verify one secondary containment access door i closed in each access opening.	24 months
SR 3.6.4.1.3	Verify each SGT subsystem can maintain $\geq$ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate $\leq$ 3000 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Dampers (SCIDs)

LCO 3.6.4.2 Each SCID shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

-----NOTES-----

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIDs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCID inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic damper, closed manual damper, or blind flange. AND	8 hours (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of recently irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	D.1 --------NOTE--------- LCO 3.0.3 is not applicable. -------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	~~AND~~	
	~~D.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND	
	D.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.2.1	Verify the isolation time of each automatic SCID is within limits.	24 months
SR 3.6.4.2.2	Verify each automatic SCID actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable in MODE 1, 2 or 3.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met. OR Two SGT subsystems inoperable in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One SGT subsystem inoperable during movement of recently irradiated fuel assemblies in the secondary containment ~~during CORE ALTERATIONS,~~ or during OPDRVs.	C.1 Restore SGT subsystem to OPERABLE status.	31 days
D. Required Action and associated Completion Time of Condition C not met.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------	
	D.1 Place OPERABLE SGT subsystem in operation.	Immediately
	<u>OR</u>	
	D.2.1 Suspend movement of recently irradiated fuel assemblies in secondary containment.	Immediately
	~~AND~~	
	~~D.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	<u>AND</u>	
	D.2.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two SGT subsystems inoperable during movement of recently irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	E.1 --------NOTE-------- LCO 3.0.3 is not applicable. -------------------- Suspend movement of recently irradiated fuel assemblies in secondary containment.	Immediately
	~~AND~~	
	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND	
	E.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for $\geq$ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months

ENCLOSURE 10

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ALTERNATIVE RADIOLOGICAL SOURCE TERM

Marked-Up Technical Specification Bases
Pages - Unit 1 (For Information Only)

Inserts Associated With Technical Specification Bases Markups

The following inserts are to be used in association with the attached Unit 1 Technical Specification Bases markups.

Insert A (Bases Page B 3.1-38)

The SLC System is also used to maintain suppression pool pH level above 7 following a loss of coolant accident (LOCA) involving significant fission product releases. Maintaining suppression pool pH levels greater than 7 following an accident ensures that iodine will be retained in the suppression pool water (Reference 2).

Insert B (Bases Page B 3.1-39)

Following a LOCA, MSLB accident, or CRD accident, offsite doses from the accident will remain within 10 CFR 50.67 limits (Reference 4) provided sufficient iodine activity is retained in the suppression pool. Credit for iodine deposition in the suppression pool is allowed (Reference 2) as long as suppression pool pH is maintained greater than 7. BSEP Alternative Source Term analyses credit the use of the SLC System for maintaining the pH of the suppression pool greater than 7.

Insert C (Bases Page B 3.3-179)

Due to radioactive decay, this function is only required to isolate secondary containment during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

Insert D (Bases Page B 3.3-187)

Also due to radioactive decay, this Function is only required to initiate the CREV System during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

Insert E (Bases Pages B 3.6-68, B 3.6-73, and B 3.6-80)

involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

Insert F (Bases Page B 3.6-69)

Due to radioactive decay, secondary containment is only required to be OPERABLE during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

Insert G (Bases Page B 3.6-74)

Due to radioactive decay, SCIVs are only required to be OPERABLE during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

Insert H (Bases Page B 3.6-80)

Due to radioactive decay, the SGT System is only required to be OPERABLE during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. In conjunction with LCOs, the limiting safety system settings, defined in LCO 3.3.1.1 as the Allowable Values, establish the threshold for protective system action to prevent exceeding acceptable limits, including this reactor vessel water level SL, during Design Basis Accidents. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of ~~10 CFR 100, "Reactor Site Criteria,"~~ limits (Ref. 2). Therefore, it is required

(continued)

10 CFR 50.67, "Accident Source Term,"

BASES

SAFETY LIMIT VIOLATIONS (continued)	to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.
REFERENCES	1. NEDE-24011-P-A (latest approved revision).
	2. ~~10 CFR 100.~~ 10 CFR 50.67.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to the UFSAR (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). Hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

10 CFR 50.67, "Accident Source Term"

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in ~~10 CFR 100, "Reactor Site Criteria"~~ (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME, Boiler and Pressure Vessel Code, 1965 Edition, including Addenda through the summer of 1967 (Ref. 5), which permits a maximum pressure

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

transient of 110% (1375 psig) of the design pressure of 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to the USAS Nuclear Power Piping Code, Section B31.1, 1967 Edition, including Addenda (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1150 psig for suction piping and 1325 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 120% of design pressures of 1150 psig for suction piping and 1325 psig for discharge piping. The most limiting of these allowances is the 110% of the RCS pressure vessel design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding the RCS pressure SL may cause RCS failure and create a potential for radioactive releases in excess of ~~10 CFR 100, "Reactor Site Criteria,"~~ 10 CFR 50.67, "Accident Source Term," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. UFSAR Section 3.1.2.2.6.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article N-910, 1965 Edition.

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.

4. ~~10 CFR 100.~~ 10 CFR 50.67.

(continued)

BASES

REFERENCES (continued)

3. NRC Safety Evaluation Report, Acceptance For Referencing of Licensing Topical Report NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17; December 27, 1987.

4. UFSAR, Section 4.3.2.5.

5. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

6. NEDO-21778-A, Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors, December 1978.

7. ASME, Boiler and Pressure Vessel Code.

8. ~~10 CFR 100.11.~~ 10 CFR 50.67

9. NEDO-21231, Banked Position Withdrawal Sequence, January 1977.

10. 10 CFR 50.36(c)(2)(ii).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram.

INSERT A

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary for both SLC pumps to inject a quantity of boron which produces a concentration of 660 ppm of natural boron in the reactor coolant at 70°F with normal reactor vessel water level. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. ~~2~~). 3 The volume versus concentration limits in Figure 3.1.7-1 and the temperature versus concentration limits in Figure 3.1.7-2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

INSERT B

The SLC System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) (Ref. ~~2~~ 5) because operating experience and probabilistic risk assessments have shown the SLC System to be important to public health and safety. Thus, it is retained in the Technical Specifications.

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions (concentration and temperature) of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path. In addition, the boron solution concentration should be within the limits of Figure 3.1.7-1 and the boron solution temperature should be within the limits of Figure 3.1.7-2.

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in the shutdown position and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Determination of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical with the analytically determined strongest control rod withdrawn. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50.62.

3 ~~2~~. UFSAR, Section 9.3.4.

5 ~~3~~. 10 CFR 50.36(c)(2)(ii).

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2. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, Final Report," February 1, 1995.

4. 10 CFR 50.67.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. The two headers are connected to a common vent line with two valves in series. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 50.67 ~~10 CFR 100~~ (Ref. 1); and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 50.67 ~~10 CFR 100~~ (Ref. 1), and adequate core cooling is maintained (Ref. 2). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.3 (continued)

need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has demonstrated these components will usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. ~~10 CFR 100.~~ 10 CFR 50.67.

2. NUREG-0803, Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping, August 1981.

3. 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Main Steam Line Isolation

1.a. Reactor Vessel Water Level—Low Level 3

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level—Low Level 3 Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level—Low Level 3 Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSLs on Level 3 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Level 3 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Level 3 Allowable Value is chosen to be the same as the ECCS Level 3 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding ~~10 CFR 100~~ 10 CFR 50.67 limits. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

This Function isolates the Group 1 valves.

1.b. Main Steam Line Pressure—Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure—Low Function is directly assumed in the analysis of the pressure regulator

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

1.b. Main Steam Line Pressure—Low (continued)

failure (Ref. 2). For this event, the closure of the MSIVs ensures that no significant thermal stresses are imposed on the RPV. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be far enough below normal turbine inlet pressures to avoid spurious isolations, yet high enough to provide timely detection of a pressure regulator malfunction.

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves except for sample line isolation valves B32-F019 and B32-F020.

1.c. Main Steam Line Flow—High

Main Steam Line Flow—High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow—High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 5). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the ~~10 CFR 100~~ limits.

10 CFR 50.67

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Reactor Vessel Water Level—Low Level 1 (continued)

limit the release of fission products. The isolation of the primary containment on Level 1 supports actions to ensure that offsite dose limits of ~~10 CFR 100~~ 10 CFR 50.67 are not exceeded. The Reactor Vessel Water Level—Low Level 1 Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level—Low Level 1 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Level 1 Allowable Value was chosen to be the same as the RPS Level 1 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

This Function isolates the Group 2, 6, and 8 valves.

2.b. Drywell Pressure—High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of ~~10 CFR 100~~ 10 CFR 50.67 are not exceeded. The Drywell Pressure—High Function, associated with isolation of the primary containment, is implicitly assumed in the UFSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Reactor Building Exhaust Radiation—High (continued)

The Reactor Building Exhaust Radiation—High signals are initiated from radiation detectors that are located in the ventilation exhaust ductwork plenum coming from the reactor building. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Two channels of Reactor Building Exhaust Radiation—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Reactor Building Exhaust Radiation—High Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, the Function is not required. In addition, the Function is also required to be OPERABLE during ~~CORE ALTERATIONS,~~ OPDRVs~~,~~ and movement of recently irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. INSERT C

ACTIONS

A Note has been provided to modify the ACTIONS related to secondary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable secondary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable secondary containment isolation instrumentation channel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Control Building Air Intake Radiation—High Function consists of two independent monitors. Two channels per trip system of Control Building Air Intake Radiation—High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude CREV System initiation. The Allowable Value was selected to ensure protection of the control room personnel.

The Control Building Air Intake Radiation—High Function is required to be OPERABLE in MODES 1, 2, and 3 and during ~~CORE ALTERATIONS,~~ OPDRVs, and movement of recently irradiated fuel assemblies in the secondary containment, to ensure that control room personnel are protected during a LOCA, fuel handling event, or vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., OPDRVs ~~CORE ALTERATIONS~~), the probability of a LOCA, main steam line break accident, or control rod drop accident ~~, or fuel damage~~ is low; thus, the Function is not required. INSERT D

ACTIONS

A Note has been provided to modify the ACTIONS related to CREV System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable CREV System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable CREV System instrumentation channel.

A.1

Because of the redundancy of sensors available to provide initiation signals and the redundancy of the CREV System design, an allowable out of service time of 7 days is provided to permit restoration of any inoperable channel to OPERABLE status. This out of service time is only acceptable provided the Control Building Air Intake Radiation—High Function is still maintaining CREV System initiation capability (refer to Required Action B.1 Bases).

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the reactor coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the reactor coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure that in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of ~~10 CFR 100~~ 10 CFR 50.67 (Ref. 1).

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the ~~10 CFR 100~~ 10 CFR 50.67 limit.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in References 2 and 3. The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the primary coolant, assumed in the Reference 3 analyses, ensure

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

that the 2 hour thyroid and whole body doses at the site boundary, resulting from an MSLB outside containment during steady state operation, will not exceed 10% of the dose guidelines of ~~10 CFR 100~~ 10 CFR 50.67.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the ~~10 CFR 100~~ 10 CFR 50.67 limits.

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems. The upper limit of 4.0 μCi/gm ensures that the thyroid dose from an MSLB will not exceed the dose guidelines of ~~10 CFR 100~~ 10 CFR 50.67 or control room operator dose limits specified in GDC 19 of 10 CFR 50, Appendix A (Ref. 5).

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least once every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of ~~10 CFR 100~~ during a postulated MSLB accident.

10 CFR 50.67

Alternatively, the plant can be placed in MODE 3 within 12 hours and in MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for placing the unit in MODES 3 and 4 are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. ~~10 CFR 100.11.~~ 10 CFR 50.67.

2. UFSAR, Section 15.6.3.

3. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, dated September 1995.

4. 10 CFR 50.36(c)(2)(ii).

5. 10 CFR 50, Appendix A, GDC 19.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.3 (continued)

administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience that has demonstrated the reliability of the explosive charge continuity.

SR 3.6.1.3.4

Verifying the isolation time of each power operated and each automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.5. The isolation time test ensures that each valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.5

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA and transient analyses. This ensures that the calculated radiological consequences of these events remain within ~~10 CFR 100~~ limits. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

10 CFR 50.67

SR 3.6.1.3.6

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. This SR includes verifying that each automatic PCIV in the Containment Atmosphere Dilution System flow path will actuate to its isolation position on the associated Group 2 and 6 primary containment isolation signals. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.1, "Primary Containment Isolation

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure. To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Dampers (SCIDs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

APPLICABLE SAFETY ANALYSES

INSERT E

There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a loss of coolant accident (LOCA) (Refs. 1 and 2) and a fuel handling accident inside secondary containment (Refs. 1 and 3). The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

(continued)

BASES (continued)

LCO

An OPERABLE secondary containment provides a control volume into which fission products that leak from primary containment, or are released from the reactor coolant pressure boundary components or irradiated fuel assemblies located in secondary containment, can be processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained, at least one door in each access to the Reactor Building must be closed, and the sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) must be OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs)~~, during CORE ALTERATIONS,~~ or during movement of recently irradiated fuel assemblies in the secondary containment. INSERT F

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 8 hours. The 8 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1~~,~~ and C.2~~, and C.3~~

Movement of recently irradiated fuel assemblies in the secondary containment~~, CORE ALTERATIONS,~~ and OPDRVs can be postulated to cause significant fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. Therefore, ~~CORE ALTERATIONS and~~ movement of recently irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable. Suspension of ~~these activities~~ this activity shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since recently irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Dampers (SCIDs)

BASES

BACKGROUND

The function of the SCIDs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Refs. 1, 2, and 3). Secondary containment isolation within the time limits specified for those isolation dampers designed to close automatically ensures that fission products that leak from primary containment following a DBA, or that are released during certain operations when primary containment is not required to be OPERABLE or take place outside primary containment, are maintained within the secondary containment boundary.

The OPERABILITY requirements for SCIDs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. These isolation devices consist of active (automatic) devices.

Automatic SCIDs close on a secondary containment isolation signal to establish a boundary for untreated radioactive material within secondary containment following a DBA or other accidents.

APPLICABLE SAFETY ANALYSES

The SCIDs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary containment boundary is required are a loss of coolant accident (Refs. 1 and 2) and a fuel handling accident INSERT E inside secondary containment (Refs. 1 and 3). The secondary containment performs no active function in response to either of these limiting events, but the boundary established by SCIDs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Maintaining SCIDs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIDs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

SCIDs form a part of the secondary containment boundary. The SCID safety function is related to control of offsite radiation releases resulting from DBAs.

The isolation dampers are considered OPERABLE when their associated accumulators are pressurized, their isolation times are within limits, and the dampers are capable of actuating on an automatic isolation signal. The dampers covered by this LCO, along with their associated stroke times, are listed in Reference 5.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, the OPERABILITY of SCIDs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIDs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs)~~, during CORE ALTERATIONS,~~ or during movement of recently irradiated fuel assemblies in the secondary containment. Moving recently irradiated fuel assemblies in the secondary containment may also occur in MODES 1, 2, and 3. INSERT G

recently

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2 ~~and D.3~~

If any Required Action and associated Completion Time are not met, the plant must be placed in a condition in which the LCO does not apply. If applicable, ~~CORE ALTERATIONS and~~ the movement of recently irradiated fuel assemblies in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since recently irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

Verifying that the isolation time of each automatic SCID is within limits, by cycling each SCID through one complete cycle of full travel and measuring the isolation time, is required to demonstrate OPERABILITY. The isolation time test ensures that the SCID will isolate in the required time period. The Frequency of this SR is once per 24 months. Operating experience has demonstrated these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.4.2.2

Verifying that each automatic SCID closes on a secondary containment isolation signal is required to minimize leakage of radioactive material from secondary containment following

(continued)

BASES

BACKGROUND (continued)

filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber beds remove gaseous elemental iodine and organic iodides, and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following an initiation signal, both SGT charcoal filter train fans start.

APPLICABLE SAFETY ANALYSES

INSERT E

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents (Refs. 2, 3, and 4). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SGT subsystem in the event of a single active failure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs)~~, during CORE ALTERATIONS,~~ or during movement of recently irradiated fuel assemblies in the secondary containment.

INSERT H

(continued)

BASES

ACTIONS (continued)

D.1, D.2.1, and D.2.2~~, and D.2.3~~

During movement of irradiated fuel assemblies, in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs, when Required Action C.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should immediately be placed in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that represent a potential for releasing a significant amount of radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk. If applicable, ~~CORE ALTERATIONS and~~ movement of recently irradiated fuel assemblies must immediately be suspended. Suspension of these activities must not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

LCO 3.0.3 is not applicable in MODE 4 or 5. However, since recently irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition D have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

E.1 and E.2~~, and E.3~~

When two SGT subsystems are inoperable, if applicable, ~~CORE ALTERATIONS and~~ movement of recently irradiated fuel assemblies in secondary containment must immediately be suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if

(continued)

BASES

ACTIONS

E.1~~,~~ and E.2~~, and E.3~~ (continued)

applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since recently irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action E.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem, by initiating (from the control room) flow through the HEPA filters and charcoal adsorbers, for ≥ 10 continuous hours ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on automatic control for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 6), except as specified in Specification 5.5.7, "Ventilation Filter Testing Program (VFTP)". The VFTP includes testing HEPA

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the main condenser. Air and noncondensible gases are collected in the main condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System for the purposes of this specification consists of the components in the following flow path from the main condenser SJAEs to the plant stack. Offgas is discharged from the main condenser via the SJAEs and diluted with steam to keep hydrogen levels below explosive concentrations. The offgas is then passed through an Offgas Recombiner System where hydrogen and oxygen are catalytically recombined into water. After recombination, the offgas is routed to an offgas condenser to remove moisture. The offgas then passes through a 30 minute delay pipe before entering the Augmented Offgas Charcoal Adsorber System. The radioactivity of the offgas recombiner effluent is monitored downstream of the offgas condenser prior to entering the 30 minute delay pipe. The Augmented Offgas Charcoal Adsorber System provides a long delay period for radioisotope decay as the offgas passes through the system. Offgas exiting the Augmented Offgas Charcoal Adsorber System is routed to the plant stack for release to the environment.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the UFSAR, Section 11.3 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the

10 CFR 50.67 limits of ~~10 CFR 100~~ (Ref. 2).

The main condenser offgas limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

(continued)

BASES

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

An alternative to Required Actions B.1 and B.2 is to place the unit in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample (taken at the discharge of the main condenser air ejector prior to dilution or discharge) to ensure that the required limits are satisfied. The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88. If the measured rate of radioactivity increases significantly (by ≥ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is indicated (by the condenser air ejector noble gas activity monitor), to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, and is acceptable, based on operating experience.

This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. Only in this condition can radioactive fission gases be in the Main Condenser Offgas System at significant rates.

REFERENCES

1. UFSAR, Section 11.3.
2. ~~10 CFR 100.~~ 10 CFR 50.67.
3. 10 CFR 50.36(c)(2)(ii).

B 3.7 PLANT SYSTEMS

B 3.7.7 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the UFSAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Section 15.7.1 (Ref. 2).

APPLICABLE SAFETY ANALYSES

10 CFR 50.67

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident (Ref. 2). A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are well below the ~~10 CFR 100~~ (Ref. 3) exposure guidelines. A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.25 (Ref. 4).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core. The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 9.1.2.
2. UFSAR, Section 15.7.1.
3. ~~10 CFR 100.~~ 10 CFR 50.67.
4. Regulatory Guide 1.25, March 23, 1972.
5. 10 CFR 50.36(c)(2)(ii).

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level

BASES

BACKGROUND

The movement of fuel assemblies or handling of control rods within the RPV requires a minimum water level of 23 ft above the top of irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level in the reactor vessel. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to well below the 10 CFR 50.67 ~~10 CFR 100~~ exposure guidelines (Ref. 3).

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies or handling of control rods, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the damaged fuel assembly rods is retained by the water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite doses are maintained well below the allowable limits of Reference 3.

RPV water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.
2. UFSAR, Section 15.7.1.
3. ~~10 CFR 100.11.~~ 10 CFR 50.67.
4. 10 CFR 50.36(c)(2)(ii).