Progress Energy

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10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: 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 Amendment
Revised Main Steam Isolation Valve Leakage Limit

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc. (PEC), is requesting a revision to the Technical Specifications (TSs) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed change revises Surveillance Requirement 3.6.1.3.9 with respect to the allowed leakage rate through each Main Steam Isolation Valve (MSIV). Specifically, the limit is revised from an allowable leakage rate of less than or equal to 11.5 scfh through each MSIV to less than or equal to 100 scfh through each main steam line (MSL) with the combined leakage of the four MSLs being less than or equal to 150 scfh. Also, to support the MSIV leakage rate change, additional automatic initiation functions for the Control Room Emergency Ventilation (CREV) system will be implemented. The associated changes to TS 3.3.7.1, "Control Room Emergency Ventilation (CREV) System Instrumentation," are also included. An evaluation of the proposed license amendment is provided in Enclosure 1.

PEC has evaluated the proposed change in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92(c), and determined that this change involves no significant hazards considerations.

PEC is providing, in accordance with 10 CFR 50.91(b), a copy of the proposed license amendment to the designated representative for the State of North Carolina.

Issuance of the requested amendment has the potential to provide significant benefit to outage activities. As such, PEC requests approval by February 1, 2006, in order to support activities for the BSEP, Unit 1 refueling outage which is currently scheduled to begin on March 4, 2006.

PEC requests that the amendment, once approved, be effective immediately, to be implemented prior to the completion of the B116R1 refueling outage (i.e., the March 2006,

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refueling outage) for Unit 1 and prior to the completion of the B216R1 (i.e., the March 2007, refueling outage for Unit 2.

Calculations supporting this amendment request contain information prepared by Applied Analysis Corporation (AAC). This information is considered to be proprietary to AAC and, as such, the calculations are not included with this submittal. Should review of supporting calculations be deemed necessary, they will be submitted separately upon the request of the NRC.

Regulatory commitments contained in this letter are identified in Enclosure 6. Please refer any questions regarding this submittal to Mr. Edward T. O'Neil, Manager - Support Services, at (910) 457-3512.

I declare, under penalty of perjury, that the foregoing is true and correct. Executed on August 11, 2005.

Sincerely,

Cornelius J. Gannon

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Enclosures:
1. Evaluation of License Amendment Request
2. Marked-up Technical Specification Page - Unit 1
3. Typed Technical Specification Page - Unit 1
4. Typed Technical Specification Page - Unit 2
5. Marked-up Technical Specification Bases Page - Unit 1 (For Information Only)
6. List of Regulatory Commitments

cc (with enclosures):

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Evaluation of License Amendment Request

Subject: Request for License Amendment
Revised Main Steam Isolation Valve Leakage Limit

1.0 Description

This letter is a request by Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc. (PEC), to amend Operating Licenses DPR-71 and DPR-62 for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2.

The proposed change revises Surveillance Requirement (SR) 3.6.1.3.9 with respect to the allowed leakage rate through each Main Steam Isolation Valve (MSIV). Specifically, the limit is revised from an allowable leakage rate of less than or equal to 11.5 scfh per MSIV to less than or equal to 100 scfh through each main steam line (MSL) with the combined leakage of the four MSLs being less than or equal to 150 scfh. Also, to support the MSIV leakage rate change, additional automatic initiation functions for the Control Room Emergency Ventilation (CREV) system will be implemented. The associated changes to TS 3.3.7.1, "Control Room Emergency Ventilation (CREV) System Instrumentation," are also included.

2.0 Proposed Change

The following change to SR 3.6.1.3.9 is proposed.

Existing SR 3.6.1.3.9	Proposed SR 3.6.1.3.9
SR 3.6.1.3.9 Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig.	SR 3.6.1.3.9 Verify leakage rate through each main steam line is ≤ 100 scfh and the combined leakage rate of all four main steam lines is ≤ 150 scfh when tested at ≥ 25 psig.

In support of the MSIV leakage rate change, additional automatic initiation functions for the CREV system will be implemented. The proposed changes to TS 3.3.7.1 follow.

Existing LCO 3.3.7.1	Proposed LCO 3.3.7.1
LCO 3.3.7.1 Two channels per trip system of the Control Building Air Intake Radiation - High Function shall be OPERABLE.	LCO 3.3.7.1 The CREV System instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE.

Existing 3.3.7.1 Applicability	Proposed 3.3.7.1 Applicability
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).	APPLICABILITY: According to Table 3.3.7.1-1
Existing Condition A	**Proposed Condition A**
A. One or more channels inoperable.	A. One or more Functions with one or more required channels inoperable.
Existing Condition B	**Proposed Condition B**
B. CREV System initiation capability not maintained.	B. One or more Functions with CREV System initiation capability not maintained.
Existing SR 3.3.7.1.3	**Proposed SR 3.3.7.1.3**
SR 3.3.7.1.3 Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 27 mR/hr.	SR 3.3.7.1.3 Perform CHANNEL CALIBRATION.

New Table 3.3.7.1-1
Control Room Emergency Ventilation (CREV) System Instrumentation

Function	Applicable Modes or Other Specified Conditions	Required Channels per Trip System	Surveillance Requirements	Allowable Value
1. Control Building Air Intake Radiation - High	1, 2, 3 (a), (b)	2	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 27 mR/hr
2. Unit [1/2] Secondary Containment Isolation - CREV Auto-Start	1, 2, 3	2	SR 3.3.6.2.2 SR 3.3.6.2.5	(c)

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

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

(c) The auto-start signal is provided from Secondary Containment Isolation logic and does not depend on a specific instrument; for Secondary Containment Isolation Instrumentation, refer to Table 3.3.6.2-1.

For convenience, Enclosure 2 contains a marked-up version of the Unit 1 TSs showing the proposed changes. Since TS Sections 3.6.1.3 and 3.3.7.1 for Unit 1 and Unit 2 are identical, only the mark-up for Unit 1 is provided. Enclosures 3 and 4 provide typed versions of the Unit 1 and

Unit 2 TSs, respectively. These typed TS pages are to be used for issuance of the proposed amendment.

PEC will make supporting changes to the TS Bases in accordance with TS 5.5.10, "Technical Specifications (TS) Bases Control Program." Enclosure 5 provides marked-up TS Bases pages for Unit 1. These pages are being submitted for information only and do not require issuance by the NRC.

3.0 Background

Reason for Change

Refurbishment of an MSIV to meet the current 11.5 scfh leakage criterion is a labor intensive effort which results in unnecessary personnel exposure and expenditure of resources. The actual leak rate observed for any one MSIV during leakage testing has also been demonstrated to be significantly influenced by a number of factors such as the MSL in which the subject MSIV is installed (i.e., the affects of asymmetric flows due to pipe configurations), actual as-installed valve stem orientation, and the temperature of the valve when the valve is closed and the leak rate testing is performed. Based on review of long-term MSIV leak rate history, a 100 scfh limit for an individual MSL and a maximum of 150 scfh for all MSLs combined leakage would significantly reduce the amount of rework required to be performed, and so would avoid incurring the exposure attendant with the current leakage criterion. The reduction in rework would also result in less wear induced by maintenance activities on the MSIVs.

Regulatory Background

This change request is based on the methodology described in NEDC-31858P-A, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems" (i.e., Reference 1). NEDC-31858P-A states that application of the alternative source term in combination with the typical boiling water reactor (BWR) MSLs and the steam flow path through the condenser box would permit plants to credit this extended release path for control of post-accident radiological releases. The report indicates that increased leakage through the MSLs up to approximately 200 scfh may be permissible with acceptable increases in control room and offsite dose consequences. The NRC has determined that NEDC-31858P-A is acceptable for direct reference for individual licensee applications, as documented in the associated NRC Safety Evaluation Report (SER) dated March 3, 1999 (i.e., Reference 2). Since BSEP Units 1 and 2 do not have Leakage Control Systems, there are no changes required in this respect and the proposed changes support revision of the MSIV leakage rate limits.

On May 30, 2002 (i.e., Reference 3), the NRC issued Amendment Nos. 221 and 246 to the Facility Operating Licenses for the BSEP, Units 1 and 2, respectively. These amendments revised the TS to replace the accident source term used in loss-of-coolant accident (LOCA), main steam line break (MSLB) accident, and control rod drop accident (CRDA) design basis analyses with an alternate source term (AST) in accordance with 10 CFR 50.67, "Accident Source Term." Section 3.0 of the safety evaluation for the AST license amendments documented the NRC's review of the alternate leakage treatment pathway. Based on the information reviewed, the NRC concluded that the main steam system piping and components which comprise the alternate leakage treatment system are seismically rugged and thus able to perform the safety function of an MSIV leakage treatment system.

4.0 Technical Analysis

There are four MSLs installed on each BSEP unit. Each MSL penetrates the primary containment and connects the nuclear steam supply to the steam turbine-generator. Each MSL is isolated by a pair of fast-closing MSIVs. These are fabricated as flow-over-seat wye-pattern globe valves with one of each pair installed just inboard of the primary containment and one outboard of the primary containment in each MSL. These valves serve to isolate the reactor coolant system in the event of steam line breaks outside primary containment, a design basis LOCA, or other events requiring containment isolation. For a detailed discussion of the system components and operating characteristics, refer to Sections 6.2.4.2 and 6.2.4.3 of the BSEP Updated Final Safety Analysis Report.

As discussed above, the NRC has concluded that the main steam system piping and components which comprise the alternate leakage treatment system (i.e. ALT path) are seismically rugged and thus able to perform the safety function of an MSIV leakage treatment system as documented in Section 3.0 of the safety evaluation for the BSEP AST license amendments (i.e., Reference 3). This request does not alter the design or function of the ALT path. Therefore, this Technical Analysis does not address the functional design of the ALT path or the capability to establish the ALT path under post-accident conditions.

Radiological Consequence Analysis

When adopting AST, BSEP did not change the existing MSIV leakage surveillance requirement to facilitate review and approval of the AST amendment and, in part, due to the originally projected impact on control room operator dose. These evaluations have been updated based on new information and approaches, including the adoption of NUREG/CR-6604, Supplement 2, dated October 2002, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation" (i.e., RADTRAD Version 3.03, Reference 4), as part of the AST analyses in place of RADTRAD Version 3.02 code used for the analysis in support of the BSEP AST amendment request (i.e., Reference 5).

With adoption of RADTRAD Version 3.03, the RADTRAD model has been modified to include an additional torus volume to account for the air space and exposed internal surface area, and by including recirculation paths between the suppression chamber air space and the modeled volume representing the drywell. The recirculation paths are given an artificially high flow in order to ensure homogeneous mixing between the two volumes is maintained. Credit for this volume depletion feature is not employed in the model during the initial two hours after event initiation during which the source term becomes fully evolved. This modeling feature has been previously approved by the NRC for use at the Fermi and Hope Creek plants (i.e., References 6 and 7).

Table 1, "Comparison of Key Radiological Analysis Assumptions," provides a comparison of the key radiological analysis assumptions reviewed and approved for the BSEP AST amendment with those used in support of the analysis to revise the MSIV leakage limits. For those that change, a brief discussion of the reason for the change is also included.

As shown in Table 1, the revised analysis credits an automatic start of one CREV filter train within two minutes of the event initiation (i.e., prior to source term release delay time-out in the model). The proposed changes to TS 3.3.7.1 incorporate the new automatic initiation functions for the CREV system.

In addition, the updated control room dose is based on recalculated direct shine contributions from the reactor building (RB) cloud, the core spray discharge piping, the Standby Gas Treatment System (SGTS) filter radiological loading, and the CREV filter radiological loading. The original evaluation credited the existing 8-inch concrete wall between the CREV filter and the control room. The updated CREV filter radiological loading calculation assumes an additional 2-inch of steel plate between the CREV filter and the control room. BSEP is committing to install shielding, consistent with the assumptions of this calculation, prior to implementation of the proposed amendment on the first BSEP unit (i.e., see Enclosure 6).

The revised direct shine calculations also use more accurate modeling of the actual geometric configurations of the radiological sources (i.e., SGTS filter, RB cloud, and core spray piping) with respect to the control room. The revised modeling better represents plant structures, equipment layout, and piping configurations. This permits crediting an additional 2-foot thick concrete wall for shielding, located in the control room interior, which was not originally included in the analysis. In addition, the revised model considers two target points for estimating dose rather than one as in the prior model. The two points represent the nearest point for an operator in the back panel area of the control room to the accident unit and the nearest point of an operator in the main control to the accident unit, respectively. The occupancy factors applied in original dose analysis calculation were unchanged for the revised calculations. However, the dose calculated at each of the two target points is combined such that the total dose value used is 25 percent of the dose calculated at the back panels position plus 100 percent of the dose calculated at the main control panels position.

This arrangement of dose analysis locations differs from the previous analysis in that the locations chosen represent actual crew work locations, whereas the previous analysis was based on evaluating the highest dose point in the control building. Because of symmetry in the control room layout, only one unit is evaluated in order to address potential accident exposure from either unit. There are no control functions or activities that take place in the area previously analyzed for the control room occupancy 30-day dose. The space is not essential to control room functioning in either normal or abnormal operating conditions. BSEP's current emergency plans post a health physics technician in the control room to monitor exposure rates.

As a result of the above enhancements, the contribution to post-LOCA control room dose from the four direct shine sources has decreased from 1.35 Rem to 0.125 Rem. Approximately half of this reduction is associated with planned upgrade of the shielding between the CREV filters and the control room; which results in a dose savings of 0.64 Rem. The balance of dose savings is primarily due to crediting the more realistic operator occupancy locations.

The analysis model prepared in support of this change request also includes a five minute increase in the assumed secondary containment positive pressure period (i.e., from five minutes to 10 minutes). This results in an analytical condition increase from three minutes to eight minutes for the period during which gap release activity leaks into secondary containment without effective negative pressure control established. This was implemented as a conservative measure to increase the analytical margin in the SGTS required performance.

The atmospheric dispersion factors (χ/Qs) are not affected by the proposed change and remain the same as those reviewed and approved for the BSEP AST amendment. Thus, the affects of atmospheric dispersion are unchanged for the purpose of calculating doses in the control room and technical support center (TSC), and at the exclusion area boundary (EAB) and low population zone (LPZ).

For conservatism, the worst case post-LOCA containment temperature profiles were referenced for the re-analysis. Section 6.2.1.1.3.2, "Primary Containment Response to Pipe Breaks," of the BSEP UFSAR provides four cases (i.e., Cases A, B, C, and D) which were used in the licensing basis to analyze the long-term pressure and temperature response of the primary containment. LOCA Case C develops a higher suppression pool peak temperature, which is more conservative for the determination of the radiological source term plate-out in the containment. The pressure profile assumptions applied in the re-analysis remain the same as those applied in the original AST analysis. Although NRC guidance (i.e., Regulatory Guide 1.183, Appendix A, Sections 3.7 and 6.2) allows for reduction of the primary containment leakage rate and MSIV leakage rate after the first 24 hours, this option is not used for the BSEP analysis. This is because BSEP has a Containment Atmosphere Control (CAC)/Containment Atmosphere Dilution (CAD) system which gradually pressurizes the primary containment following the initial pressure reduction. The previous analysis found that assuming steady MSIV leakage at 25 psig bound this evolution of pressure reduction and increase. Therefore, as a simplifying assumption it is retained in the revised analysis. Also, consistent with the original AST analysis, the primary containment leakage rate is conservatively set to 80 percent of the initial 24-hour leak rate value for the remainder of the 30-day duration.

The results of the DBA-LOCA re-analysis demonstrate acceptable control room, TSC, and offsite doses with a single MSL leakage limit of less than or equal to 100 scfh and the combined leakage of the four MSLs being less than or equal to 150 scfh.

PEC has also re-analyzed the dose associated with NUREG-0737, Item II.B.2, post-accident vital area access and determined the impact from the increased MSIV leakage rates. Mission doses were evaluated and they continue to meet the dose criteria of NUREG-0737, Item II.B.2.

The following table compares the current doses with those analyzed for increased MISV leakage.

	Control Room (Rem)	**TSC (Rem)**	**EAB (Rem)**	**LPZ (Rem)**	**Mission Dose (Rem)**
Current dose	3.62	1.15	0.64	1.36	4.45
Dose assuming increased MSIV leakage limits	4.39	1.69	2.37	4.08	4.94
Regulatory limit	5.00	5.00	25.00	25.00	5.00

Control Room Emergency Ventilation Initiation Analysis

The CREV system is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. Two independent CREV subsystems are each capable of fulfilling the stated safety function. Currently, the CREV system instrumentation has two trip systems, either of which can initiate the CREV system; however, the BSEP CREV system design only permits one CREV train to be in service at a time. Each trip system receives input from the two Control Building Air Intake Radiation - High Function channels. The Control Building Air Intake Radiation - High Function is arranged in a one-out-of-two logic for each trip system. The CREV system initiates to pressurize the control room to minimize the consequences of radioactive material in the control room environment. The LOCA analysis performed in support of the BSEP AST license amendments assumed operator action to start one CREV filter train at 20 minutes after initiation of the event. In support of the MSIV leakage rate change, the revised analysis assumes an automatic start of one CREV filter train within the first two minutes from event initiation, which is before the initial gap release activity reaches the ambient environment.

The required automatic initiation of the CREV system will be accomplished by using signals from the secondary containment isolation logic as an input to each division of the CREV control logic. As shown in Table 3.3.6.2-1, "Secondary Containment Isolation Instrumentation," of the BSEP TSs, secondary containment isolates on (1) Reactor Vessel Water Level - Low Level 2, (2) Drywell Pressure - High, or (3) Reactor Building Exhaust Radiation - High. Reactor Vessel

Water Level - Low Level 2 and Drywell Pressure - High provide primary indication of a LOCA. Therefore, the automatic CREV actuation will be accomplished using secondary containment isolation logic associated with these two functions. CREV automatic actuation will not occur as a result of secondary containment isolations due to Reactor Building Exhaust Radiation - High signals. The existing Control Building Air Intake - Radiation High function remains and will continue to provide operator protection for non-LOCA events.

The Reactor Vessel Water Level - Low Level 2 and the Drywell Pressure - High signals are both designed as emergency core cooling system control logic inputs with two divisions using a one-out-of-two taken twice logic arrangement that provides an input to each secondary containment isolation control logic division. Thus, the proposed automatic CREV initiation, using signals from the secondary containment isolation logic, provides redundant/diverse protection for control room operators in the event of a LOCA, the event of concern with respect to revising the existing MSIV leakage requirements.

The basis for crediting the CREV train start within the first two minutes of the postulated event is the nominal response time of the secondary containment isolation instrument logic compared to the assumed availability of AC power to the CREV system fans. The LOCA parameters from which secondary containment isolation actuates, Reactor Vessel Water Level - Low Level 2 and Drywell Pressure - High, will occur within the first second of a design basis LOCA. The logic will respond in no more than a few seconds to initiate CREV system damper repositioning, which starts the CREV system fans. The longest potential delay of initiation is from an assumed loss of offsite power through the emergency diesel generator start/load sequence, which is analytically 15 seconds from event initiation until emergency busses are powered. Thus, the redundant secondary containment isolation logic signals to auto-start the CREV trains will be present when AC power is restored at 15 seconds from event initiation, and well within the assumed two minutes of the revised analysis (i.e., before fuel gap release initially reaches the external environment).

As previously discussed, the analysis model prepared in support of this change request includes a five minute increase in the assumed secondary containment positive pressure period. The positive pressure period duration is independent of the assumption of CREV start within two minutes since the secondary containment isolation logic signals are diverse and are redundant for each division of CREV trip system logic. Therefore, both CREV trip system divisions retain full capability to start the preferred CREV train based on a valid one-out-of-two signal.

The required logic modifications will be performed such that faults originating in the CREV logic cannot affect either the secondary containment isolation logic or the Reactor Vessel Water Level - Low Level 2 and Drywell Pressure - High functions themselves. The existing TS Limiting Conditions for Operation (LCOs) for TS 3.3.6.2, "Secondary Containment Isolation Instrumentation," are more restrictive than those required for CREV system instrumentation. As such, operability of the automatic CREV initiation using signals from the secondary containment isolation logic as an input is appropriately ensured.

Based on the above discussion, it is concluded that the proposed automatic CREV initiation, using signals from the secondary containment isolation logic, provides redundant/diverse protection for control room operators in the event of a LOCA. Reactor Vessel Water Level - Low Level 2 and Drywell Pressure - High provide primary indication of the potential of a LOCA and, as such, will automatically initiate the CREV system via the secondary containment isolation logic in a timely manner, consistent with the assumptions of the revised LOCA analysis. Existing and proposed TS LCO requirements ensure operability of the instrumentation.

5.0 Regulatory Safety Analysis

5.1 No Significant Hazards Consideration

PEC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change revises SR 3.6.1.3.9 with respect to the allowed leakage rate through each MSIV. Specifically, the limit is revised from an allowable leakage rate of less than or equal to 11.5 scfh per MSIV to less than or equal to 100 scfh for any one MSL with the combined leakage of the four MSLs being less than or equal to 150 scfh. Also, to support the MSIV leakage rate change, additional automatic initiation functions for the CREV system will be implemented. The associated changes to TS 3.3.7.1, "Control Room Emergency Ventilation (CREV) System Instrumentation," are also made.

   The proposed change to the MSIV leakage limit does not involve physical change to any plant structure, system, or component. As a result, no new failure modes of the MSIVs has been introduced. The CREV system initiation logic is being modified; however, this system performs a mitigating function and has no impact on any initiating event frequency. Therefore, the proposed changes cannot increase in the probability a previously evaluated accident.

   A plant-specific radiological analysis has been performed to assess the effects of the proposed increase in MSIV leakage acceptance criteria in terms of offsite doses and control room doses. The analysis shows the dose contribution from the proposed increase in leakage acceptance criteria is acceptable compared to dose limits prescribed in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and 10 CFR 50.67(b)(2)(iii) for control room personnel. The CREV system initiation logic modification will result in automatic initiation of the CREV system based on signals from the secondary containment isolation logic as an input to each division of the CREV control logic. This change is made to ensure that doses to control room personnel remain within the requirements of 10 CFR 50.67(b)(2)(iii) in the event of a loss-of-coolant-accident.

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

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change to the MSIV leakage limit will not adversely impact MSIV functionality and will not create a failure of the MSIVs of a different kind than previously considered. The CREV system initiation logic is being modified to initiate automatically using signals from the secondary containment isolation logic. This provides redundant/diverse protection for control room operators in the event of a LOCA. The required logic modifications will be performed such that faults originating in the CREV logic cannot affect either the secondary containment isolation logic or the functions which initiate secondary containment isolation.

   Based on the above, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The allowable leak rate specified for the MSIVs is used to quantify a maximum amount of leakage assumed to bypass containment. The results of the re-analysis supporting these changes were evaluated against the dose limits contained in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and 10 CFR 50.67(b)(2)(iii) for control room personnel. Sufficient margin relative to the regulatory limits is maintained even when conservative assumptions and methods are utilized. The CREV system initiation logic is being modified to initiate automatically using signals from the secondary containment isolation logic. This provides redundant/diverse protection for control room operators in the event of a LOCA.

   Based on the above, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, PEC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The BSEP design was reviewed for construction under the "General Design Criteria for Nuclear Power Plant Construction" issued for comment by the Atomic Energy Commission in July 1967 and is committed to meet the intent of the General Design Criteria (GDC), published in the Federal Register on May 21, 1971 as Appendix A to 10 CFR Part 50. Criterion 19, "Control room," requires that the control room have adequate radiation protection to permit access and occupancy under accident conditions without personnel exceeding regulatory limits. Additionally, radiation exposure to the public must not exceed 10 CFR 50.67 limits.

The radiological consequences of the design basis LOCA at Brunswick have been evaluated against the dose limits contained in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and 10 CFR 50.67(b)(2)(iii) for control room personnel and it has been determined that, considering the proposed changes, sufficient margin relative to the regulatory limits is maintained even when conservative assumptions and methods are utilized.

Based on the considerations discussed above: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 Environmental Considerations

10 CFR 51.22(c)(9) identifies certain licensing and regulatory actions, which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 References

1. BWROG letter, W. G. Warren to NRC, dated November 22, 1999, "Transmittal of Approved GE Licensing Topical Report, NEDC-31858P-A," dated August 1999.

2. NRC letter to the BWROG, F. M. Akstulewicz to T. A. Green, dated March 3, 1999, "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."

3. Letter from Brenda L. Mozafari (U.S. NRC) to J. S. Keenan, "Brunswick Steam Electric Plant, Units 1 And 2 - Issuance of Amendment Re: Alternative Source Term," dated May 30, 2002, ADAMS Accession Number ML021480483.

4. NUREG/CR-6604, Supplement 2, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation," dated October 2002.

5. Letter from John S. Keenan to the U. S. Nuclear Regulatory Commission (Serial: BSEP 01-0063), "Request for License Amendments - Alternative Radiological Source Term," dated August 1, 2001, ADAMS Accession Number ML012180234.

6. NRC Letter from Mr. David P. Beaulieu to Mr. William T. O'Connor, Jr., "Fermi 2 - Issuance of Amendment RE: Selective Implementation of Alternative Radiological Source Term Methodology," dated September 28, 2004, ADAMS Accession Number ML042430179.

7. NRC Letter from Mr. George F. Wunder to Mr. Roy A. Anderson, "Hope Creek Generating Station - Issuance of Amendment RE: Containment Requirements During Fuel Handling and Removal of Charcoal Filters," dated April 15, 2003, ADAMS Accession Number ML030760293.

8. Letter from Cornelius J. Gannon to U. S. Nuclear Regulatory Commission (Serial: BSEP 04-0093), "Response to Generic Letter 2003-01, Control Room Habitability," dated July 29, 2004, ADAMS Accession Number ML042170286.

<u>Precedence</u>

This change request is based on the methodology described in NEDC-31858P-A, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," (i.e., Reference 1). NEDC-31858P-A states that application of the alternative source term in combination with the typical boiling water reactor (BWR) MSLs and the steam flow path through the condenser box would permit plants to credit this extended release path for control of post-accident radiological releases. The report indicates that increased leakage through the MSLs up to approximately 200 scfh may be permissible with acceptable increases in control room and offsite dose consequences. The NRC has determined that NEDC-31858P-A is acceptable for direct reference for individual licensee applications, as documented in the associated NRC Safety Evaluation Report (SER) dated March 3, 1999 (i.e., Reference 2).

On May 30, 2002 (i.e., Reference 3), the NRC issued Amendment Nos. 221 and 246 to the Facility Operating Licenses for the BSEP, Units 1 and 2, respectively. These amendments revised the TS to replace the accident source term used in LOCA, MSLB accident, and CRDA design basis analyses with an alternate source term in accordance with 10 CFR 50.67, "Accident Source Term."

Table 1 - Comparison of Key Radiological Analysis Assumptions			
Parameter	**AST Value**	**Current Value**	**Discussion**
Reactor power	2981 MWt (i.e., 102% of 2923 MWt, licensed thermal power for BSEP)	2981 MWt	
Dose conversion factors	FGR11 and FGR12	FGR11 and FGR12	
Control room volume	298,650 ft^3	298,650 ft^3	
Control room isolation	Manual 20 minutes	Automatic < 2 minutes	This modification will result in control room isolation prior to source term release from secondary containment. The proposed changes to TS 3.3.7.1 incorporate the new automatic initiation functions for the CREV system.
Control room normal ventilation makeup flow	2100 cfm	2100 cfm	
Control room filtered makeup flow	1500 cfm	1500 cfm	
Control room filtered recirculation flow	400 cfm	400 cfm	
Control room filter efficiency	Aerosol 95% Elemental 90% Organic 90%	Aerosol 95% Elemental 90% Organic 90%	

Table 1 - Comparison of Key Radiological Analysis Assumptions

Parameter	AST Value	Current Value	Discussion
Control room unfiltered inleakage	0, 3000, and 10,000 cfm	2,000 cfm	This assumption is based on the control building tracer gas test results submitted to the NRC (i.e., Reference 8) in response to Generic Letter 2003-01.
Control room breathing rate	3.4E-04 m^3/sec	3.4E-04 m^3/sec	
Control room occupancy factors	0 - 24 hours 1.0 1 - 4 days 0.6 4 - 30 days 0.4	0 - 24 hours 1.0 1 - 4 days 0.6 4 - 30 days 0.4	
Core inventory	Calculated by ORIGEN	Calculated by ORIGEN	
Onset of gap release phase	2.0 minutes	2.0 minutes	
Core release fractions and timing - drywell atmosphere	RG 1.183, Table 1	RG 1.183, Table 1	
Core release fractions and timing - ECCS leakage	Duration (hrs) <u>0.5</u> <u>1.5</u> Iodine: 0.05 0.25	Duration (hrs) <u>0.5</u> <u>1.5</u> Iodine: 0.05 0.25	
Iodine species fraction Particulate/aerosol Elemental Organic	<u>Atm</u> <u>Supp Pool</u> 95 0 4.85 100 0.15 0	<u>Atm</u> <u>Supp Pool</u> 95 0 4.85 100 0.15 0	
Drywell volume	164,000 ft^3	164,000 ft^3	

Table 1 - Comparison of Key Radiological Analysis Assumptions			
Parameter	**AST Value**	**Current Value**	**Discussion**
Torus free-air volume	Not included	122,000 ft^3	This assumption change credits torus volume depletion after the initial 2 hours of an event. See Enclosure 1, Section 4.0 for additional discussion.
Containment release after 24 hours	0.5 %/day 0.4 %/day	0.5 %/day 0.4 %/day	
MSIV leakage (total)	46 scfh	150 scfh	The assumed MSL/MSIV leak rate is increased from 46 scfh (i.e., the current 11.5 scfh per valve limit yields a total combined allowed leakage of 46 scfh for the four MSLs) to 150 scfh. The 150 scfh is partitioned into two paths of 100 scfh and 50 scfh for conservatism in determining the reduction credit for the radiological source term in the steam lines due to plate-out depletion.
Duration of release	30 days	30 days	
Reactor Building (RB) positive pressure period (PPP)	5 minutes	10 minutes	This assumption is incorporated by increasing the period during which gap release activity leaks into secondary containment, without effective negative pressure control, from three to eight minutes in the analysis model. Although this results in a negative impact on control room dose, the RB PPP was increased

Table 1 - Comparison of Key Radiological Analysis Assumptions			
Parameter	**AST Value**	**Current Value**	**Discussion**
			to provide SGTS performance analytical margin.
Drywell natural deposition	10% Powers Model	10% Powers Model	
Main steam line deposition model	Brockmann-Bixler	Well Mixed Volume	Updated based on the adoption of the RADTRAD Version 3.03
Pressure (0-30 days) Temperature (0-30 days)	4.33 atm 560 degrees F	4.33 atm (i.e., 49 psig) Variable.	The pressure assumed for the analysis of main steam line deposition is the same as the previous AST analysis. Since flow rates are fixed in the RADTRAD models, this assumption has no significant affect on the deposition model. See Enclosure 1, Section 4.0 for additional discussion regarding modeling of containment response. Key aspects of the model, relevant to the alternative release path depletion credit, were verified as a good engineering practice. In particular, the physical dimensions of the path and the cool down rate for steam piping were reviewed. The pipe runs forming the alternate release pathways were verified against the current approved configuration drawings and change postings. The cool down rate was calculated using an analytical simulation model prepared in the GOTHIC thermal-

Table 1 - Comparison of Key Radiological Analysis Assumptions			
Parameter	**AST Value**	**Current Value**	**Discussion**
			hydraulic code. This calculation replaces the conservative assumption that the steam piping remained at 560 degrees F during the analysis period.
Condenser deposition elemental & particulate efficiency • Main steam primary drain line path • Main steam alternate drain line path	 99.8% 99,6%	 99.8% 99,6%	
SGTS filter efficiency, all species	99%	99%	
ECCS leak rate (includes 20x multiplier)	20 gpm	20 gpm	
Duration of release	30 days	30 days	
Suppression pool liquid volume	86,450 ft^3	86,450 ft^3	

Marked-up Technical Specification Page - Unit 1

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).~~

According to Table 3.3.7.1-1.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each channel.

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

The CREV System Instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE.

SURVEILLANCE REQUIREMENTS

---NOTE---

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains CREV initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Perform CHANNEL CALIBRATION. ~~The Allowable Value shall be ≤ 27 mR/hr.~~	24 months
SR 3.3.7.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

New Table 3.3.7.1-1

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation (CREV) System Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Control Building Air Intake Radiation - High	1, 2, 3 (a), (b)	2	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 27 mR/hr
2. Unit 1 Secondary Containment Isolation - CREV Auto-Start	1, 2, 3	2	SR 3.3.6.2.2 SR 3.3.6.2.5	(c)

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

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

(c) The auto-start signal is provided from Secondary Containment Isolation logic and does not depend on a specific instrument; for Secondary Containment Isolation Instrumentation, refer to Table 3.3.6.2-1.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.7	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.8	Remove and test the explosive squib from each shear isolation valve of the TIP System.	In accordance with the Inservice Testing Program
SR 3.6.1.3.9	~~Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig~~.	In accordance with the Primary Containment Leakage Rate Testing Program

Verify leakage rate through each main steam line is ≤ 100 scfh and the combined leakage rate of all four main steam lines is ≤ 150 scfh when tested at ≥ 25 psig.

Typed Technical Specification Page - Unit 1

3.3 INSTRUMENTATION

3.3.7.1 Control Room Emergency Ventilation (CREV) System Instrumentation

LCO 3.3.7.1 The CREV System Instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.7.1-1.

ACTIONS

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

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

SURVEILLANCE REQUIREMENTS

--NOTE--
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains CREV initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.7.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation (CREV) System Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Control Building Air Intake Radiation - High	1, 2, 3 (a), (b)	2	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 27 mR/hr
2. Unit 1 Secondary Containment Isolation - CREV Auto-Start	1, 2, 3	2	SR 3.3.6.2.2 SR 3.3.6.2.5	(c)

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

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

(c) The auto-start signal is provided from Secondary Containment Isolation logic and does not depend on a specific instrument; for Secondary Containment Isolation Instrumentation, refer to Table 3.3.6.2-1.

3.3 INSTRUMENTATION

3.3.7.2 Condenser Vacuum Pump Isolation Instrumentation

LCO 3.3.7.2 Four channels of the Main Steam Line Radiation—High Function for condenser vacuum pump isolation shall be OPERABLE.

APPLICABILITY: MODES 1 and 2 with a condenser vacuum pump in service.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status. <u>OR</u> A.2 -----NOTE----- Not applicable if inoperable channel is the result of an inoperable condenser vacuum pump trip breaker or isolation valve. ---------- Place channel or associated trip system in trip.	12 hours 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR Condenser vacuum pump isolation capability not maintained.	B.1	Isolate condenser vacuum pumps.	12 hours
	OR		
	B.2	Isolate main steam lines.	12 hours
	OR		
	B.3	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains condenser vacuum pump isolation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.7.2.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.7.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.2.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 6 x background.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.7.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including condenser vacuum pump trip breaker and isolation valve actuation.	24 months

3.3 INSTRUMENTATION

3.3.8.1 Loss of Power (LOP) Instrumentation

LCO 3.3.8.1 The LOP instrumentation for each Function in Table 3.3.8.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,

When the associated diesel generator is required to be OPERABLE by LCO 3.8.2, "AC Sources—Shutdown."

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Place channel in trip.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Declare associated diesel generator (DG) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

--NOTES--

1. Refer to Table 3.3.8.1-1 to determine which SRs apply for each LOP Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 2 hours provided: (a) for Function 1, the associated Functions maintains initiation capability for three DGs; and (b) for Function 2, the associated Function maintains DG initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.8.1.1	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.8.1.2	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.8.1.3	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.8.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.8.1-1 (page 1 of 1)
Loss of Power Instrumentation

FUNCTION	REQUIRED CHANNELS PER BUS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)			
a. Bus Undervoltage	1	SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 3115 V and ≤ 3400 V
b. Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 0.5 seconds and ≤ 2.0 seconds
2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)			
a. Bus Undervoltage	3	SR 3.3.8.1.1 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 3706 V and ≤ 3748 V
b. Time Delay	3	SR 3.3.8.1.1 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 9.0 seconds and ≤ 11.0 seconds

3.3 INSTRUMENTATION

3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring

LCO 3.3.8.2 Two RPS electric power monitoring assemblies shall be OPERABLE for each inservice RPS motor generator set or alternate power supply.

APPLICABILITY: MODES 1 and 2,

MODES 3, 4, and 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or both inservice power supplies with one electric power monitoring assembly inoperable.	A.1	Remove associated inservice power supply(s) from service.	72 hours
B. One or both inservice power supplies with both electric power monitoring assemblies inoperable.	B.1	Remove associated inservice power supply(s) from service.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1 or 2.	C.1	Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B nct met in MODE 3, 4, or 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.	D.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.8.2.1	-----------------------------NOTE----------------------------- Only required to be performed prior to entering MODE 2 from MODE 3 or 4, when in MODE 4 for ≥ 24 hours. --- Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.8.2.2	Perform CHANNEL CALIBRATION for each RPS motor generator set electric power monitoring assembly. The Allowable Values shall be: a. Overvoltage ≤ 129 V. b. Undervoltage ≥ 105 V. c. Underfrequency ≥ 57.2 Hz.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.8.2.3	Perform CHANNEL CALIBRATION for each RPS alternate power supply electric power monitoring assembly. The Allowable Values shall be: a. Overvoltage ≤ 132 V. b. Undervoltage ≥ 108 V. c. Underfrequency ≥ 57.2 Hz.	24 months
SR 3.3.8.2.4	Perform a system functional test.	24 months

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.1.3.6	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.7	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.8	Remove and test the explosive squib from each shear isolation valve of the TIP System.	In accordance with the Inservice Testing Program
SR 3.6.1.3.9	Verify leakage rate through each main steam line is ≤ 100 scfh and the combined leakage rate of all four main steam lines is ≤ 150 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

Typed Technical Specification Page - Unit 2

3.3 INSTRUMENTATION

3.3.7.1 Control Room Emergency Ventilation (CREV) System Instrumentation

LCO 3.3.7.1 The CREV System Instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.7.1-1

ACTIONS

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

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

SURVEILLANCE REQUIREMENTS

--NOTE--

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains CREV initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.7.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation (CREV) System Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Control Building Air Intake Radiation - High	1, 2, 3 (a), (b)	2	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 27 mR/hr
2. Unit 2 Secondary Containment Isolation - CREV Auto-Start	1, 2, 3	2	SR 3.3.6.2.2 SR 3.3.6.2.5	(c)

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

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

(c) The auto-start signal is provided from Secondary Containment Isolation logic and does not depend on a specific instrument; for Secondary Containment Isolation Instrumentation, refer to Table 3.3.6.2-1.

3.3 INSTRUMENTATION

3.3.7.2 Condenser Vacuum Pump Isolation Instrumentation

LCO 3.3.7.2 Four channels of the Main Steam Line Radiation—High Function for condenser vacuum pump isolation shall be OPERABLE.

APPLICABILITY: MODES 1 and 2 with a condenser vacuum pump in service.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more channels inoperable.	A.1	Restore channel to OPERABLE status.	12 hours
	OR		
	A.2	---------------NOTE------------ Not applicable if inoperable channel is the result of an inoperable condenser vacuum pump trip breaker or isolation valve. --------------------------------- Place channel or associated trip system in trip.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR Condenser vacuum pump isolation capability not maintained.	B.1	Isolate condenser vacuum pumps.	12 hours
	OR		
	B.2	Isolate main steam lines.	12 hours
	OR		
	B.3	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

--NOTE--

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains condenser vacuum pump isolation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.7.2.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.7.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.2.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 6 x background.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.7.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including condenser vacuum pump trip breaker and isolation valve actuation.	24 months

3.3 INSTRUMENTATION

3.3.8.1 Loss of Power (LOP) Instrumentation

LCO 3.3.8.1 The LOP instrumentation for each Function in Table 3.3.8.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,

When the associated diesel generator is required to be OPERABLE by LCO 3.8.2, "AC Sources—Shutdown."

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Place channel in trip.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Declare associated diesel generator (DG) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

NOTES

1. Refer to Table 3.3.8.1-1 to determine which SRs apply for each LOP Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 2 hours provided: (a) for Function 1, the associated Functions maintains initiation capability for three DGs; and (b) for Function 2, the associated Function maintains DG initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.8.1.1	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.8.1.2	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.8.1.3	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.8.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.8.1-1 (page 1 of 1)
Loss of Power Instrumentation

FUNCTION	REQUIRED CHANNELS PER BUS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)			
a. Bus Undervoltage	1	SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 3115 V and ≤ 3400 V
b. Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 0.5 seconds and ≤ 2.0 seconds
2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)			
a. Bus Undervoltage	3	SR 3.3.8.1.1 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 3706 V and ≤ 3748 V
b. Time Delay	3	SR 3.3.8.1.1 SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 9.0 seconds and ≤ 11.0 seconds

3.3 INSTRUMENTATION

3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring

LCO 3.3.8.2 Two RPS electric power monitoring assemblies shall be OPERABLE for each inservice RPS motor generator set or alternate power supply.

APPLICABILITY: MODES 1 and 2,

MODES 3, 4, and 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or both inservice power supplies with one electric power monitoring assembly inoperable.	A.1	Remove associated inservice power supply(s) from service.	72 hours
B. One or both inservice power supplies with both electric power monitoring assemblies inoperable.	B.1	Remove associated inservice power supply(s) from service.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1 or 2.	C.1	Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met in MODE 3, 4, or 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.	D.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.8.2.1	------NOTE------ Only required to be performed prior to entering MODE 2 from MODE 3 or 4, when in MODE 4 for ≥ 24 hours. ------------ Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.8.2.2	Perform CHANNEL CALIBRATION for each RPS motor generator set electric power monitoring assembly. The Allowable Values shall be: a. Overvoltage ≤ 129 V. b. Undervoltage ≥ 105 V. c. Underfrequency ≥ 57.2 Hz.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.8.2.3	Perform CHANNEL CALIBRATION for each RPS alternate power supply electric power monitoring assembly. The Allowable Values shall be: a. Overvoltage ≤ 132 V. b. Undervoltage ≥ 108 V. c. Underfrequency ≥ 57.2 Hz.	24 months
SR 3.3.8.2.4	Perform a system functional test.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.7	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.8	Remove and test the explosive squib from each shear isolation valve of the TIP System.	In accordance with the Inservice Testing Program
SR 3.6.1.3.9	Verify leakage rate through each main steam line is ≤ 100 scfh and the combined leakage rate of all four main steam lines is ≤ 150 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

Marked-up Technical Specification Bases Page - Unit 1
(For Information Only)

B 3.3 INSTRUMENTATION

B 3.3.7.1 Control Room Emergency Ventilation (CREV) System Instrumentation

BASES

BACKGROUND

The CREV System is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. Two independent CREV subsystems are each capable of fulfilling the stated safety function. The instrumentation and controls for the CREV System automatically initiate action to pressurize the main control room (MCR) to minimize the consequences of radioactive material in the control room environment.

Insert 1

In the event of a Control Building Air Intake Radiation—High signal, the CREV System is automatically started in the radiation/smoke protection mode. Air is then recirculated through the charcoal filter, and sufficient outside air is drawn in through the normal intake to maintain the MCR slightly pressurized with respect to outside atmosphere.

Insert 2

The CREV System instrumentation has two trip systems, either of which can initiate the CREV System. Each trip system receives input from the ~~two Control Building Air Intake Radiation—High Function channels.~~ The Control Building Air Intake Radiation—High Function is arranged in a one-out-of-two logic for each trip system. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a CREV System initiation signal to the initiation logic.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ability of the CREV System to maintain the habitability of the MCR is explicitly assumed for the design basis accident as discussed in the UFSAR safety analyses (Refs. 1 and 2). CREV System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents, does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A.

CREV System instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

(continued)

Insert 1

In the event of a loss of coolant accident (LOCA), the Unit 1 Secondary Containment Isolation - CREV Auto-Start signal will automatically start the CREV System in the radiation/smoke protection mode. The automatic CREV actuation is generated from the portion of the secondary containment isolation logic associated with the Reactor Vessel Water Level - Low Level 2 and the Drywell Pressure - High. CREV automatic actuation will not occur as a result of secondary containment isolations due to Reactor Building Exhaust Radiation - High signals. The Control Building Air Intake - Radiation High function provides protection for non-LOCA events.

Insert 2

Functions listed above. The Reactor Vessel Water Level - Low Level 2 and the Drywell Pressure - High signals are both designed as emergency core cooling system control logic inputs with two divisions using the one-out-of-two taken twice logic arrangement that provides an input to each secondary containment isolation control logic division. Thus, the automatic CREV initiation, using signals from the secondary containment isolation logic, provides redundant/diverse protection for control room operators in the event of a LOCA.

individual instrumentation channel Functions specified in Table 3.3.7.1-1.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The OPERABILITY of the CREV System instrumentation is dependent upon the OPERABILITY of the ~~Control Building Air Intake Radiation—High instrumentation channel Function~~. The Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

(5)

(5)

CREV System Isolation Function specified in the Table.

Allowable Values are specified for ~~Control Building Air Intake Radiation—High Function~~. Trip setpoints are specified in the setpoint calculations. The setpoints are selected to ensure that the trip settings do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setting less conservative than the trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setting is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., control building air intake radiation), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are determined from the analytic limits, corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for calibration based errors. These calibration based instrument errors are limited to instrument drift, errors associated with measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Control Building Air Intake Radiation - High

The control building air intake radiation monitors measure radiation levels in the control building air intake plenum. A high radiation level may pose a threat to MCR personnel; thus, automatically initiating the CREV System.

(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 OPDRVs and movement or 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), the probability of a LOCA, main steam line break accident, or control rod drop accident is low; thus, the Function is not required. 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 3

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

(continued)

Insert 3

2. Unit 1 Secondary Containment Isolation - CREV Auto-Start

Unit 1 Secondary Containment Isolation - CREV Auto-Start provides post-LOCA operator protection. The Secondary Containment Isolation Instrumentation automatically initiates closure of appropriate secondary containment isolation dampers (SCIDs), starts the Standby Gas Treatment (SGT) System, and provides CREV Auto-Start (Refer to B 3.3.6.2-1 for details associated with Secondary Containment Isolation Instrumentation) on (1) Reactor Vessel Water Level - Low Level 2, (2) Drywell Pressure - High, or (3) Reactor Building Exhaust Radiation - High. Since Reactor Vessel Water Level - Low Level 2 and Drywell Pressure - High provide primary indication of a LOCA, only secondary containment isolations resulting from these signals provide CREV Auto-Start. The Reactor Vessel Water Level - Low Level 2 and the Drywell Pressure - High signals are both designed as emergency core cooling system control logic inputs with two divisions using the one-out-of-two taken twice logic arrangement that provides an input to each secondary containment isolation control logic division. Thus, automatic CREV initiation, using signals from the secondary containment isolation logic, provides redundant/diverse protection for control room operators in the event of a LOCA.

The Allowable Values for the Secondary Containment Isolation Instrumentation are provided in LCO 3.3.6.2.

Unit 1 Secondary Containment Isolation - CREV Auto-Start is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected in the event of a LOCA. Unit 1 Secondary Containment Isolation - CREV Auto-Start is not dependent upon OPERABILITY of the Reactor Building Exhaust Radiation - High function.

BASES

ACTIONS

A.1 (continued)

affected

acceptable provided the ~~Control Building Air Intake Radiation—High~~ Function is still maintaining CREV System initiation capability (refer to Required Action B.1 Bases). If the Function is not maintaining CREV System initiation capability, Condition B must be entered.

If the inoperable channel cannot be restored to OPERABLE status within the 7 day allowable out of service time, one CREV subsystem must be placed in the radiation/smoke protection mode of operation per Required Action A.1. The method used to place the CREV subsystem in operation must provide for automatically re-initiating the subsystem upon restoration of power following a loss of power to the CREV subsystem. Placing one CREV subsystem in the radiation/smoke protection mode of operation provides a suitable compensatory action to ensure that the automatic radiation protection function of the CREV System is not lost.

B.1

affected

Required Action B.1 is intended to ensure that appropriate action is taken if multiple, inoperable, untripped channels result in the ~~Control Building Air Intake Radiation—High~~ Function not maintaining CREV System initiation capability. The Function is considered to be maintaining CREV System initiation capability when sufficient channels are OPERABLE or in trip such that one trip system will generate an initiation signal for one CREV subsystem from the Function on a valid signal. ~~For the Control Building Air Intake Radiation—High Function,~~ this would require one trip system to have one channel OPERABLE or in trip. With CREV System initiation capability not maintained, one CREV subsystem must be placed in the radiation/smoke protection mode of operation per Required Action B.1 to ensure that control room personnel will be protected in the event of a Design Basis Accident. The method used to place the CREV subsystem in operation must provide for automatically re-initiating the subsystem upon restoration of power following a loss of power to the CREV subsystem.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.8

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.9

2, 5, 13, and 14

The analyses in References ~~2 and 5~~ are based on leakage that is less than the specified leakage rate. ~~Leakage through each MSIV must be ≤ 11.5 scfh when tested at ≥ P_t (25 psig). The MSIV leakage rate must be verified to be in accordance with the leakage test requirements of the Primary Containment Leakage Rate Testing Program.~~ The Primary Containment Leakage Rate Testing Program has been established in accordance with 10 CFR 50.54(o) to implement the requirements of 10 CFR Part 50, Appendix J, Option B (Ref. 6), and conforms with Regulatory Guide 1.163 (Ref. 7) and Nuclear Energy Institute (NEI) 94-01 (Ref. 8) except for the following:

a. BNP may use standard glass tube and ball type flowmeters with an accuracy of 5% of full scale. This is an exception to the flowmeter accuracy requirements of ANSI/ANS 56.8-1994 (Ref. 9) referenced in NEI 94-01 (Ref. 8), Section 8.0. The basis for this exception is described in Reference 10.

(continued)

Leakage through each main steam line must be ≤ 100 scfh when tested at ≥ P_t (25 psig). The combined leakage rate for all four main steam lines must be ≤ 150 scfh when tested at ≥ 25 psig in accordance with the Primary Containment Leakage Rate Testing Program.

BASES

REFERENCES (continued)

11. NRC SER, Brunswick 1 & 2 - Amendments No. 10 and 36 to Operating Licenses Revising Technical Specifications to Grant Exemptions from Specific Requirements of 10 CFR 50 Appendix J, dated November 8, 1977.

12. NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," June 2000.

13. NRC letter, Brunswick Steam Electric Plant, Units 1 and 2 - Issuance of Amendment Re: Alternative Source Term, May 30, 2002.

14. BNP Calculation No. BNP-RAD-007, Rev. 2, DBA-LOCA Radiological Dose With Alternate Source Term.

BASES

BACKGROUND (continued)

The CREV System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem whole body dose or its equivalent to any part of the body. A single CREV subsystem will slightly pressurize the control room to prevent infiltration of air from surrounding buildings. CREV System operation in maintaining control room habitability is discussed in the UFSAR, Sections 6.4 and 9.4, (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the CREV System to maintain the habitability of the control room is an explicit assumption for the design basis accident presented in the UFSAR (Ref. 3). The radiation/smoke protection mode of the CREV System is assumed (explicitly or implicitly) to operate following a loss of coolant accident, fuel handling accident, main steam line break, and control rod drop accident. The radiological doses to control room personnel as a result of a DBA are summarized in Reference 3. Postulated single active failures that may cause the loss of outside or recirculated air from the control room are bounded by BNP radiological dose calculations for control room personnel.

The CREV System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

Two redundant subsystems of the CREV System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA if unfiltered leakage into the control room is > ~~10,000~~ 2,000 cfm.

The CREV System is considered OPERABLE when the individual components necessary to support the radiation protection mode are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Emergency recirculation fan is OPERABLE;

b. HEPA filter and charcoal adsorber bank are not excessively restricting flow and are capable of performing their filtration and adsorption functions; and

(continued)

List Of Regulatory Commitments

The following table identifies those actions committed to by Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc. (PEC) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to the Manager - Support Services at the Brunswick Steam Electric Plant (BSEP).

Commitment	Schedule
1. The updated Control Room Emergency Ventilation (CREV) filter radiological loading calculation assumes an additional 2-inch of steel plate between the CREV filter and the control room. BSEP will install shielding, consistent with the assumptions of this calculation.	Prior to implementation of the proposed amendment on the first BSEP unit.