

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
RELATED TO AMENDMENT NOS. 246 AND 274  
TO RENEWED FACILITY OPERATING LICENSES NOS. DPR-71 AND DPR-62  
CAROLINA POWER & LIGHT COMPANY  
BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2  
DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated January 22, 2007, as supplemented on June 21, July 18, July 31, and October 15, 2007, and January 24, February 14, March 5, and March 21, 2008, the Carolina Power & Light Company (the licensee) requested amendments to Renewed Operating Licenses DPR-71 and DPR-62 for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2, respectively. The proposed amendments would change the BSEP Technical Specifications (TSs) to support the transition to AREVA fuel and core design methodologies.

BSEP is a General Electric (GE) boiling water reactor (BWR) BWR/4 design. BSEP is currently operating at 120 percent of originally licensed thermal power at extended power uprate (EPU) and Maximum Extended Load Line Limit Analysis (MELLLA) conditions. BSEP currently uses GE14 fuel and GE methodologies for performing safety analyses and developing the core operating limits report (COLR). The licensee plans to transition BSEP from GE fuel and methodologies to AREVA fuel methodologies. The first transition core will be Unit 1 in the spring of 2008, when the licensee will load 248 fresh AREVA ATRIUM-10 fuel bundles. The second transition cycle is planned to have additional fresh ATRIUM-10 fuel such that most of the central bundles are ATRIUM-10 with remaining GE14 bundles located in the core periphery. The Unit 1 amendment is planned for implementation prior to startup from the spring 2008 refueling outage and the Unit 2 amendment is planned for implementation prior to startup from the spring 2009 refueling outage.

The supplements dated June 21, July 18, July 31, and October 15, 2007, and January 24, February 14, and March 5, and March 21, 2008, provided additional information that clarified the application, did not expand the scope of the original *Federal Register* notice, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 4, 2007 (72 FR 68208).

2.0 REGULATORY EVALUATION

The Nuclear Regulatory Commission (NRC) staff reviewed the license amendment request (LAR) to evaluate the applicability of the AREVA methodologies to BSEP, to confirm that the use of the methodologies is within the NRC approved ranges, and to verify that the results of the

analyses are in compliance with the requirements of the following General Design Criteria (GDC) specified in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.

- GDC-10, "Reactor design," requiring the reactor design (reactor core, reactor coolant system (RCS), control and protection systems) to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs),
- GDC-12, "Suppression of reactor power oscillations," requiring that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible, or can be reliably and readily detected and suppressed,
- GDC-15, "Reactor coolant pressure boundary," requiring the RCS and associated auxiliary, control, and protection systems to be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs,
- GDC-26, "Reactivity control system redundancy and capability," requiring two independent reactivity control systems of different design principles be provided, one of which is capable of holding the reactor subcritical under cold conditions,
- GDC-27, "Combined reactivity control system capability," requiring the reactivity control systems to be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions,
- GDC-28, "Reactivity limits," requiring the reactivity control systems to be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core.
- GDC-35, "Emergency core cooling," requiring a system to provide abundant emergency core cooling to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts.

### 3.0 TECHNICAL EVALUATION

The licensee's original application (Reference 1) requested changes to the TSs to support the fuel transition, but did not include information to justify the use of AREVA fuel or methodologies at BSEP. In particular, the licensee did not include information to demonstrate the applicability of the previously-approved AREVA Topical Reports (TRs) to BSEP or to show that the acceptance criteria are met.

In general, methodologies or computer codes used to support licensing basis analyses are documented in TRs which are reviewed by the NRC staff on a generic basis. In the NRC staff safety evaluation (SE) approving the TR, the staff defines the basis for acceptance in conjunction with any limitations and conditions on use of the TR, as appropriate. A generic TR describing a methodology or computer code does not provide the full justification for each plant-specific application. In situations where a plant-specific LAR references a TR that has not been previously applied, the staff requests that the licensee submit a plant-specific analysis to demonstrate applicability of the TR.

By References 2 through 5, the licensee submitted information to demonstrate compliance with the staff limitations and conditions imposed for application of the TRs, and to demonstrate the applicability of the AREVA codes and methods for BSEP at EPU conditions. The licensee performed analyses of the affected licensing basis events, based on the AREVA methods, to demonstrate that the use of the codes and methods is within the approved ranges and that the results of the analyses meet the applicable acceptance criteria.

The NRC staff has reviewed the LAR (Reference 1) in conjunction with the supplemental information (References 2 through 5), and the responses to the staff's requests for additional information (RAIs) (References 6, 7, and 36), to (1) evaluate the acceptability of the BSEP transition to AREVA ATRIUM-10 fuel, (2) evaluate the use of the associated AREVA methodologies for licensing applications, and (3) confirm adequate technical basis for the proposed TS changes. In addition, the staff visited the AREVA site in Richland, WA from July 30 to August 2, 2007, to review the BSEP-specific safety analyses and associated fuel methodologies.

### 3.1 Applicability of AREVA Methodologies

#### 3.1.1 Computer Codes and Methods Used

As indicated in Reference 8, the licensee performs licensing analyses with a suite of AREVA codes and methods described below. The NRC staff evaluated the applicability of these codes and methods specifically to BSEP operating at EPU conditions.

XCOBRA/XCOBRA-T (XN-NF-84-105-P-A): XCOBRA predicts the steady-state thermal-hydraulic performance of BWR cores at various operating conditions and power distributions. It is used to evaluate pressure drops, channel and bypass flow distributions, and minimum critical power ratios (MCPRs), as well as the hydraulic compatibility of fuel designs. XCOBRA-T predicts the transient thermal-hydraulic performance of BWR cores during postulated system transients and is used to evaluate the change in critical power ratio ( $\Delta$ CPR) for the limiting fuel bundles in the core. As documented in XN-NF-84-105-P-A (Reference 16), the XCOBRA-T code has been approved by the NRC for use in BWR licensing applications. The use of the steady-state XCOBRA code has been accepted by the NRC staff (Reference 17) based on approval of XCOBRA-T and the similarity of the thermal hydraulic models between the codes. The BSEP licensing analysis (discussed in Section 3.2 of this SE) shows that the core thermal-hydraulic conditions during steady state and transient conditions are within the NRC-approved range of the code. In addition, in Reference 8, the licensee demonstrated compliance with the SE restrictions associated with XN-NF-84-105-P-A. Therefore, the staff concludes that the application of XCOBRA and XCOBRA-T for the BSEP core thermal-hydraulic calculations is acceptable.

SPCB (Siemens Power Corporation BWR) Critical Power Correlation (EMF-2209-P-A): The safety limit minimum critical power ratio (SLMCPR) is imposed to protect at least 99.9 percent of the fuel rods in the core from boiling transition during steady state and transient conditions. For BSEP, planned to be loaded with ATRIUM-10 fuel, the licensee used the SPCB critical power correlation, which has been approved by the NRC for ATRIUM-10 fuel (Reference 18). The BSEP licensing analysis (discussed in Section 3.2 below) shows that the operating conditions during steady state and transient conditions are within the NRC-approved range of the correlation, and applicable operating and safety limits will be met. Therefore, the staff concludes that the use of the SPCB correlation is acceptable for ATRIUM-10 fuel.

With the introduction of the ATRIUM-10 fuel, the BSEP cores will contain co-resident GE14 fuel during the transition cycles. Therefore, acceptable critical power performance of the co-resident fuel must be established. The licensee proposes to apply the NRC-approved TR, EMF-2245-P-A (Reference 19) to model the critical power performance of the co-resident fuel. The SE restriction in Reference 19 requires that technology transfer to licensees who may be responsible for using these processes will be accomplished through Siemens Power Corporation (currently AREVA) and licensee procedures consistent with the requirements of Generic Letter (GL) 83-11, Supplement 1. GL 83-11, Supplement 1, stipulates the performance of an independent benchmarking calculation by AREVA for comparison to licensee-generated results in order to verify that the application of AREVA critical heat flux correlations is properly applied for the first application by a licensee. AREVA, in Reference 8, states that the SE restriction is implemented in their work practices. The NRC staff finds that the SE restriction is adequately addressed and the application of EMF-2245-P-A to evaluate the critical power performance of the co-resident fuel is acceptable.

COTRANSA2 (Advanced Nuclear Fuel (ANF) ANF-913-P-A): COTRANSA2 is a BWR system transient analysis code with models representing the reactor core, reactor vessel, steam lines, recirculation loops, and control systems. It is used to evaluate key reactor system parameters during core-wide BWR transient events. These parameters, such as power, flow, pressure, and temperature, are provided as boundary conditions to the hot channel analyses for  $\Delta$ CPR determination. As documented in ANF-913-P-A, the code has been generically approved by the NRC to analyze system responses to fast transients in BWRs (Reference 20).

The licensee proposes to use COTRANSA2 to perform the system analysis of the following fast AOO and anticipated transient without scram (ATWS) events: (1) load rejection with no bypass, (2) turbine trip with no bypass, (3) feedwater controller failure max demand, (4) pressure regulator downscale failure, and (5) ATWS main steam isolation valve closure and pressure regulator failure open. In Reference 8, the licensee demonstrated compliance to each restriction in the staff's approving SE. The staff finds that the code provides reasonable results and the BSEP plant specific analysis (discussed in Section 3.2 below) shows that the code is applied within the NRC-approved range. Therefore, the staff concludes that the licensee's use of COTRANSA2 in performing analysis of fast transient events for the BSEP is acceptable.

CASMO-4/MICROBURN-B2 (EMF-2158-P-A): The two principal computer programs for BWR nuclear design and analysis used by AREVA are CASMO-4 and MICROBURN-B2. The CASMO-4 code is a two dimensional multi group transport theory code used to calculate the lattice physics constants of BWR fuel assemblies. The MICROBURN-B2 code is a two group nodal code used for the three-dimensional simulation of the nuclear and thermal-hydraulic conditions in BWR cores. The MICROBURN-B2 code determines core-wide nodal neutron flux, fission power, and coolant density distributions, reactivity parameters, nodal exposure and nuclide density distributions, channel inlet flow distributions, and fuel thermal performance parameters such as linear heat generation rate (LHGR), axial planar linear heat generation rate, and critical power ratio (CPR). These results are used to design fuel cycles, to assess safety margins, and to monitor operating reactor cores.

As documented in EMF-2158P-A (Reference 21), the NRC has previously approved both CASMO-4 and MICROBURN-B2 in calculating the nuclear characteristics. However, to assess the applicability of EMF-2158-P-A to BSEP operation at EPU, the NRC staff evaluated the technical basis related to the CASMO-4 lattice spectrum/depletion code and the corresponding core simulator code, MICROBURN-B2, benchmarking and validation extension for the EPU conditions. In Reference 12, the licensee provided detailed technical information to

demonstrate that BSEP operating parameters at EPU conditions are within the applicability ranges of the staff's original review of EMF-2158-P-A. Specifically, the licensee concluded that the analyses demonstrate the following:

1. The steady state neutronic and thermal hydraulic analytical methods and code systems used to perform the safety analyses supporting the EPU conditions are being applied within the NRC-approved applicability ranges, and
2. The assessment database and the assessed uncertainty of models used in all of the licensing codes remain valid and applicable for the EPU conditions.

#### *Bundle Operating Conditions*

AREVA's analytical methods and code systems have been previously used for applications with assembly powers comparable to the EPU conditions expected at BSEP. The AREVA neutronic methodologies are characterized by technically rigorous treatment of phenomena and are well benchmarked with more than 100 cycles of operation in addition to gamma scan data for ATRIUM-10. Recent operating experience is tabulated in Table 7.1 and Table 7.2 of Reference 12. These tables present the reactor operating conditions and, in particular, the average and hot assembly powers for both U.S. and European applications. As can be seen from this information, the average and peak bundle powers in this experience base exceed those associated with BSEP at EPU conditions.

The increased steam flow from power uprate comes from increased power in normally lower power assemblies in the core, operating at higher power levels. High powered assemblies in uprated cores will be subject to the same LHGR, maximum axial planar linear heat generation rate (MAPLHGR), MCPR, and cold shutdown margin limits and restrictions as high powered assemblies in non uprated cores. The similarity of operating conditions between the original licensed power level and EPU conditions assures that the neutronic methods used to compute the nodal reactivity and power distributions remain valid. Furthermore, the neutronic characteristics computed by the steady state simulator and used in safety analysis remain valid.

Reference 12 provides additional information showing how BSEP, at EPU conditions, compares with a sample of BWRs for which AREVA has previously been or is currently the fuel vendor. The data demonstrates that the core thermal power, maximum bundle power, and maximum exit void fraction for BSEP at EPU are comparable to other uprated plants operating in the U.S. and in Europe.

#### *Applicability of Biases and Uncertainties*

During the week of July 30, 2007, the NRC staff reviewed and audited the results of the analysis to determine the reliability of the methods to predict EPU conditions in the core. The licensee provided a detailed description of the validation that has been performed to qualify the computer codes and analysis methods that are used for the nuclear design and analysis of BWRs. The audit information describing the neutronics methodology benchmarking process was provided to staff in Reference 14. Additional information regarding BSEP specific traversing incore probe (TIP) validation data was provided in Reference 7 and is evaluated in conjunction with the SLMCPR methodology in Section 3.1.2 of this SE. AREVA engages in an on going qualification program, which includes experimental and numerical comparisons (TIP comparisons, gamma scanning, and calculations using the MCNP code), to confirm the continued applicability of the pin power and assembly power uncertainties, and to confirm, on a continuing basis, the

acceptability of power distribution predictions with the CASMO-4/MICROBURN-B2 code system. The NRC staff finds that AREVA's existing benchmark data is supported by gamma scan comparisons, which include both 9x9 and 10x10 AREVA fuel geometries. The database specifically includes the ATRIUM-10 design, which will be used for BSEP. The gamma scan comparisons show good agreement between calculated and measured Barium-140 density distributions for both radial and axial values. The staff, therefore, has reasonable assurance that the pin and radial power uncertainties will remain applicable for introduction of ATRIUM-10 at BSEP. Additional staff discussion regarding the applicability of the radial power uncertainty based on BSEP specific EPU TIP data is provided in Section 3.1.2 of this SE under SLMCPR.

As a part of the review of the AREVA steady state methods for BSEP, the NRC staff evaluated the applicability of the void prediction methods. For nuclear design, AREVA uses the [REDACTED]' void correlation to determine the void fraction. In Table B-1 of Reference 12, the licensee provided the extent of the AREVA multi-rod void fraction database, specifically covering the ATRIUM-10 design and expected EPU operating conditions. Figure B-2 and Figure B-3 of Reference 12 provide comparisons of predicted versus measured void fractions for the AREVA multi-rod void fraction validation database using the [REDACTED] correlation. These figures show that the predictions reliably fall within  $\pm 0.05$  (predicted – measured) error bands with little bias, and that there is no observable trend of uncertainty as a function of void fraction. The [REDACTED] void correlation database also included the FRIGG data which was based on a circular array of 36 rods with a central thimble. The [REDACTED] void-quality correlation was qualified by AREVA against both the FRIGG void measurements and ATRIUM-10 measurements. In response to RAI Number 3, the licensee stated (Reference 6) that despite the significantly different geometrical configurations between FRIGG and ATRIUM-10, the behavior of the [REDACTED] calculations, when compared to the measured data, is remarkably similar, as illustrated in Figure 3.2 of Reference 6. This similarity of results indicates that the [REDACTED] void-quality correlation is applicable to a range of geometries larger than the differences between ATRIUM-10 and GE14 and, thus, is equally applicable to the GE14 fuel design. Based on the staff's review of the void quality correlation benchmarking data, the staff finds that the void quality correlation used in nuclear design is applicable for the BSEP LAR.

The NRC staff reviewed AREVA's topical report EMF-2158-P-A, in conjunction with additional information provided by the licensee (References 6, 7, and 12) and gathered during audits (Reference 14), to ensure that the steady-state neutronic methodology described in topical report EMF-2158-P-A is applicable for use in BSEP at EPU conditions. As a part of the audit, the staff reviewed several calculational workbooks and computer code outputs, and was briefed collectively and in one-on-one sessions on all of the above stated subjects. Based upon the information provided, the staff finds that AREVA analytical methods and the associated uncertainties and biases are acceptable for application to BSEP at EPU conditions provided the exit void fractions do not exceed the current validated range. Should the void fraction exceed the current validated range, the approved methodology requires AREVA to submit additional justification for NRC staff review for possible increases in power distribution uncertainties, or the need for additional validation data to support the use of current void fraction correlation and cross-section methods.

Other Methods for Neutronic Safety Analyses (XN-NF-80-19-P-A): The CASMO-4/MICROBURN-B2 code system is also used to perform other neutronic safety

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<sup>1</sup> The information in [REDACTED] contained proprietary information and as such has been redacted in this nonproprietary version of SE.

analyses. In this LAR, the licensee proposes to use this code system to analyze the following: (1) cold shutdown margin, (2) standby liquid control system shutdown margin, (3) control rod withdrawal error event, (4) loss of feedwater heating event, (5) control rod drop accident, (6) fuel assembly mislocation event, and (7) fuel assembly misorientation event. The NRC staff has approved the CASMO-4/MICROBURN-B2 methods for these applications in XN-NF-80-19-P-A (Reference 22) or in subsequent approved TRs (References 21 and 23) superseding portions of XN-NF-80-19-P-A.

Since the CASMO-4/MICROBURN-B2 methods are approved by the NRC and the licensee applied the methods based on BSEP plant specific analyses, the licensee's proposed application of these methods is acceptable.

SLMCPR Methodology (ANF-524-P-A): The SLMCPR is imposed to protect at least 99.9 percent of the fuel rods in the core from boiling transition during steady state and transient conditions. The NRC approved the TR ANF-524-P-A, which identifies the fuel and non-fuel related uncertainties and the statistical process used to determine a MCPR safety limit. In Reference 8, the licensee demonstrated compliance to each restriction in the staff's SE approving TR ANF-524-P-A. As discussed in further detail in Section 3.1.2 of this SE, the staff finds that the appropriate values of the MICROBURN-B2 uncertainties are used for the BSEP SLMCPR. Therefore, the staff finds the licensee's use of the SLMCPR methodology as documented in ANF-524-P-A in support of the BSEP licensing application acceptable.

Stability Methodology (NEDO-32465-A): BSEP has implemented the Boiling Water Reactor Owners Group (BWROG) Long Term Solution (LTS) Option III as their licensing basis stability protection methodology (Reference 25). To support Option III, the licensee uses the RAMONA5-FA system analysis code to generate the Delta over Initial CPR versus Oscillation Magnitude (DIVOM) relationship, which provides the CPR performance during reactor instability (Reference 26). RAMONA5-FA is able to calculate the limiting bundle response in terms of power, flow, CPR, and other parameters in the time domain. The RAMONA5-FA code is currently under generic review at the NRC. Therefore, acceptability of the application of the RAMONA5-FA method is determined on a plant-specific basis. The NRC staff reviewed the Brunswick Unit 1 Cycle 17 Option III stability analysis based in part on the RAMONA5-FA code and found the results to be reasonable. Based on NRC review of previous and current applications of the RAMONA5-FA code for DIVOM analysis, including numerous site audits, the NRC finds that application of the RAMONA5-FA code to BSEP licensing application is acceptable.

Backup Stability Protection (BSP) analyses are performed in anticipation of the long term Option III solution becoming unavailable. BSP is a prevention approach where certain areas on the power-to-flow map, where instability is likely based on decay ratio calculations, are excluded from operation. The calculations to support BSP are performed using the NRC-approved STAIF code (References 27 and 28). In Reference 8, the licensee demonstrated compliance to each restriction in the staff's SE approving use of the STAIF code. Therefore, the staff finds that the licensee's use of the STAIF code in support of BSEP licensing application is acceptable.

EXEM BWR-2000 LOCA Methodology (EMF-2361-P-A): The AREVA methodology for showing compliance with 10 CFR Part 50, Appendix K, is referred to as the EXEM BWR-2000 Evaluation Model. This model was reviewed and approved by the NRC staff in Reference 29. The EXEM BWR-2000 methodology employs three primary codes. The reactor system and hot channel response is evaluated with RELAX (Reference 29), fuel assembly heatup during the loss-of-coolant accident (LOCA) is analyzed with HUXY (Reference 30), which incorporates

approved cladding swelling and rupture models (Reference 31), and stored energy and fuel characteristics are determined with RODEX2 (Reference 32). In Reference 8, the licensee demonstrated compliance to each restriction in the staff's SEs approving these codes. Therefore, the staff finds that the licensee's use of the EXEM BWR-2000 and associated code systems in support of BSEP licensing application is acceptable.

#### Codes and Methods Summary:

In Reference 8, the licensee evaluated its compliance with the restrictions specified in each of the staff's SEs approving the AREVA TRs applied in the LAR. Accordingly, the staff concludes that the licensee adequately demonstrated conformance to the SE conditions. The licensee performed a plant-specific demonstration analysis of the limiting licensing basis events with the AREVA codes and methods to show that the use of those codes and methods is acceptable. The demonstration analysis (discussed in Section 3.2 of this SE) shows that the results are reasonable and meet the applicable acceptance criteria. Therefore, the staff concludes that the application of the NRC-approved AREVA codes and methods to the BSEP for licensing analysis is acceptable.

#### 3.1.2 Transition Core Approach

The BSEP Units have 560-bundle cores. The Unit 1 Cycle 17 core will contain 248 fresh ATRIUM-10 bundles and 312 GE14 bundles (250 once-burnt, 59 twice-burnt, 3 thrice-burnt). The NRC staff reviewed the licensee's approach to address the technical issues associated with mixed-core operation, including hydraulic compatibility, critical power performance, and impact on core design and licensing analyses.

#### Hydraulic Compatibility:

In Reference 9, the licensee reported the results of the hydraulic compatibility analysis to demonstrate that the hydraulic flow resistance of the reload ATRIUM-10 bundles will be sufficiently similar to the legacy GE14 bundles in the reactor such that there is no significant degradation in total core flow or unfavorable distribution of the flow among the bundles. Applicable criteria have been reviewed and approved by the NRC in TR XN-NF-80-19-P-A, Volume 4, Revision 1, and ANF-89-98-P-A, Revision 1, and Supplement 1 (References 33 and 34). The mixed core was analyzed with the XCOBRA steady state thermal-hydraulic code. This code was used to analyze the thermal-hydraulic performance of the mixed core configuration. The calculations were performed with explicit modeling of the ATRIUM-10 and GE14 geometries and covering several power-to-flow conditions, including rated and offrated conditions. Top-, mid-, and bottom-peaked axial power shapes were considered, and results were provided for both average and hot bundles (radial peaking of 1.5).

The licensee reported that [[

]] The staff concludes that licensee's hydraulic compatibility analysis provides reasonable assurance that introduction of ATRIUM-10 bundles will not significantly impact the core flow distribution.

#### Critical Power Correlation:



The CPR is evaluated for each fuel type in the core using calculated local fluid conditions and an appropriate critical power correlation. Fuel-type-specific correlation coefficients for ATRIUM-10 fuel are based on NRC-approved SPCB correlation (Reference 18). For the co-resident GE14 fuel, an NRC-approved process described in Reference 19 is used to develop the correlation coefficients and the associated uncertainty for application of the SPCB critical power correlation to the co-resident fuel. If adequate test data is available for the co-resident fuel, the correlation coefficients and uncertainty are determined using the process consistent with the approval of the correlation. When test data for the co-resident fuel is not available to AREVA, fuel-type-specific correlation coefficients and uncertainty are developed using calculated CPR data provided by the utility based on an alternate or an "indirect" approach described in Reference 19. For BSEP, the latter indirect approach, consistent with the NRC approval in Reference 19, was used to develop correlation coefficients and uncertainty based on CPR data provided to AREVA by the licensee. The NRC staff finds the application of the critical power correlation to co-resident fuel acceptable.

#### SLMCPR:

The Unit 1 Cycle 17 safety limit MCPR analysis uses the SPCB critical power correlation additive constants and additive constant uncertainty for ATRIUM-10 fuel and co-resident GE14 fuel as described above. The SLMCPR analysis explicitly addresses the mixed core effects and is performed each cycle using the cycle-specific core configuration. Each fuel type present in the core is explicitly modeled using appropriate geometric data, thermal hydraulic characteristics, and power distribution information (from CASMO-4 and MICROBURN-B2 analyses). CPR is evaluated for each assembly using fuel type specific SPCB correlation coefficients. Plant and fuel type specific uncertainties are considered in the statistical analysis performed to determine the SLMCPR.

As discussed in Section 3.1.1 of this SE under CASMO-4/MICROBURN-B2, the staff evaluated the impact of the proposed introduction of ATRIUM-10 fuel on the power distribution uncertainties used in safety analyses. In order to demonstrate the applicability of currently assumed radial power uncertainty value in SLMCPR calculations, the licensee provided BSEP-specific TIP data over five operating cycles at or near rated EPU power level (in Reference 7). The database includes 73 full core gamma TIP measurements: 43 spanning Unit 1 Cycles 14, 15, and 16 (i.e., through August 2006), and 30 spanning Unit 2 Cycles 16 and 17 (i.e., to October 2006). The BSEP specific database is comprised of the deviation between the calculated radial TIP and the measured radial TIP versus core power, core-power-to-flow ratio, and core average void fraction. The dataset covering the full range of expected EPU conditions clearly demonstrates that the D-lattice radial bundle power uncertainty reported in the Reference 21 topical report is conservative for BSEP. BSEP SLMCPR analyses are based on the radial bundle power uncertainty value of  $[[ \quad \quad \quad ]]$  reported in Reference 21, rather than the BSEP specific value of  $[[ \quad \quad \quad ]]$ . Therefore the BSEP specific value is conservative relative to the topical report value by  $[[ \quad \quad \quad ]]$ , due primarily to BSEP implementation of gamma TIPs for local power range monitor (LPRM) calibration. The licensee further showed that by reducing the correlation coefficient by 50 percent, the BSEP specific value increases by  $[[ \quad \quad \quad ]]$  and the value would still maintain a margin of  $[[ \quad \quad \quad ]]$ . The actual radial uncertainty value used for BSEP SLMCPR analysis is  $[[ \quad \quad \quad ]]$  to account for TIP and LPRM out of service conditions.

The licensee stated that the BSEP gamma TIP measurement database based on GE14 and GE13 is applicable to ATRIUM-10 fuel because the database, which includes measurements of

a full core of GE13 (9x9), a mixed core of GE13 and GE14, and a full core of GE14 (10x10), shows changes in fuel design have no significant impact on CASMO-4/MICROBURN-B2 uncertainties.

Based on above, the NRC staff finds that the currently approved radial uncertainties for the D lattice, used for the SLMCPR, are reasonable and that the SLMCPR approach for the projected BSEP operation, including transition cores, is acceptable.

#### Core Design and Licensing Analysis:

For core design and nuclear safety analyses, the neutronic cross-section data is developed for each fuel type in the core using CASMO-4. Geometric and nuclear design data (e.g., enrichment distribution) that is required to prepare CASMO-4 input for the co-resident fuel is obtained from a third party (i.e., the licensee) under a proprietary agreement. MICROBURN-B2 is used to design the core and provide input to safety analyses (core neutronic characteristics, power distributions, etc.). Each fuel assembly is explicitly modeled in MICROBURN-B2 using cross-section data from CASMO-4 and geometric data appropriate for the fuel design.

Fuel assembly thermal mechanical limits for both ATRIUM-10 and co-resident fuel are verified and monitored for each mixed core designed by AREVA. The thermal mechanical limits established by the vendor of the co-resident fuel are applied for mixed (transition) cores. The licensee provided AREVA with the thermal mechanical limits (steady-state and transient) for the co-resident fuel. AREVA performed design and licensing analyses to demonstrate that the core design meets the limits during steady-state and AOO conditions. The NRC staff finds the approach acceptable.

#### Transition Core Approach Summary:

Based on above, the NRC staff finds that the AREVA methodology is being used in accordance with NRC approval to perform design and licensing analyses for mixed cores. The cycle design and licensing analyses explicitly consider each fuel type in mixed core configurations. Co-resident fuel input parameters are being developed consistent with the methodology and, in general, are developed in the same manner as for AREVA fuel. Limits are established for each fuel type and operation within these limits is verified by the monitoring system during operation. Therefore, the staff concludes that the licensee's transition core approach is acceptable.

### 3.2 Plant Specific Licensing Analysis

#### 3.2.1 AOOs

The plant responses to the limiting AOOs are analyzed for each reload cycle. The results are used to establish the operating limit minimum critical power ratio (OLMCPR) and to confirm that the American Society of Mechanical Engineers (ASME) overpressure criterion is met. To support the current LAR, the licensee performed a reload transient analysis to cover the projected operating conditions within the licensed power-to-flow map, equipment out of service options, and scram speed options (i.e., TS-required scram speed and nominal scram speed). For the initial application of AREVA fuel and methodology to BSEP, the reload analysis consisted of simulation of [[ ]] transient events to cover the rated and off-rated operating conditions. The results were used to determine the OLMCPR limits for ATRIUM-10 and co-resident GE14 fuel.

The OLMCPRs are established so that less than 0.1 percent of the fuel rods in the core are expected to experience boiling transition during an AOO initiated from rated or off-rated conditions. For Unit 1, the operating limits are based on the TSs two-loop operation SLMCPR of 1.11 and a single-loop operation SLMCPR of 1.12. The transient  $\Delta$ CPR is added to the SLMCPR to determine the OLMCPR for that transient event. The limiting event sets the OLMCPR for plant operation. The AOO events analyzed for OLMCPR include: load rejection with no bypass, turbine trip with no bypass, feedwater controller failure maximum demand, loss of feedwater heating, rod withdrawal error, and slow recirculation increase events.

The NRC staff finds the licensee determined the OLMCPR based on the approach that was accepted by the staff. In addition, a review of the Unit 1 Cycle 17 reference core design in conjunction with the reload AOO analyses indicates a Maximum Fraction of Limiting Critical Power Ratio (MFLCPR) of 0.93, providing a 7 percent margin to the OLMCPR at the most limiting exposure point in the cycle. This represents a slight increase in design margin from Unit 1 Cycle 16, which had a MFLCPR of 0.95. The MCPR thermal margin information for Unit 1 Cycle 17 provides an indication that, with the proposed transition in fuel and methodology, cores will continue to be designed in a manner such that CPR margins will not be adversely impacted.

In addition, the licensee confirmed in a response to a staff RAI that MCPR impact due to the Part 21 report (Part 21 Report 2007-29-00) is accounted for in the operating limits presented in support of this LAR (Reference 15). The Part 21 report discussed an error in the local power peaking distribution in the ATRIUM-10 test assemblies used to determine the critical power performance in the BWR Karlstein Thermal Hydraulic (KATHY) test loop. The NRC staff finds this acceptable.

ASME overpressure analysis is performed to demonstrate compliance with the ASME Boiler and Pressure Vessel Code. The analysis also demonstrates that the plant-specific safety/relief valve configuration has sufficient capacity and performance to prevent the reactor vessel pressure from reaching the safety limit of 110 percent of the design pressure. Main steam isolation valve (MSIV) closure and turbine stop valve closure (without bypass) events were analyzed with the COTRANSA2 systems code assuming 102 percent rated power at both 99 percent and 104.5 percent rated core flow at the end of cycle (EOC) conditions. The maximum pressure of 1352 pounds per square inch gauge (psig) occurred in the lower plenum for the MSIV closure event at the 104.5 percent core flow condition. The results demonstrated that the maximum vessel pressure limit of 1375 psig is not exceeded. The NRC staff finds that ASME overpressure protection is adequately demonstrated.

The staff finds that: (1) the licensee used approved codes and methodologies to perform the AOO analyses; (2) the values used for the input parameters are appropriate in predicting the consequences; (3) the calculated responses of key reactor and system parameters are reasonable; and (4) the results of the analysis show that the fuel integrity and RCS pressure boundary integrity acceptance criteria will be met. Therefore, the staff concludes that the impact of the proposed LAR on the AOOs is acceptable.

### 3.3.2 Thermal-hydraulic Stability

BSEP is currently operating under the requirements of the reactor stability LTS Option III approved by the NRC staff in GE Licensing Topical Report NEDO-32465-A (Reference 25). The BSP solution is used when the oscillation power range monitor (OPRM) is inoperable. The OPRM system is designed to provide for an automatic scram when the system detects power

oscillations above the setpoint. The licensee stated that the OPRM hardware at BSEP will remain consistent with the assumptions in the hot channel oscillation magnitude (HCOM) calculation of record and, therefore, HCOM does not need to be recalculated to support this LAR.

The staff reviewed NEDO-32465-A and determined that the HCOM portion of the Option III calculation is based on hardware-specific items such as the LPRM assignments and the reactor protection system (RPS) trip logic. The NRC staff therefore finds that the HCOM portion of the Option III calculation is indeed hardware specific and need not change as a result of the proposed LAR.

The OPRM system is armed only when plant operation is within the Option III trip-enabled region where the plant is susceptible to instability. Setpoints for the OPRM system are determined in a two-step process that is based on the MCPR. The MCPR margin that exists prior to the onset of oscillations is determined for two scenarios: a two recirculation pump trip from full power at the highest rod line, and steady-state operation at 45 percent core flow with the core at the OLMCPR. From these MCPR values, the change in CPR during an oscillation is assessed to determine the DIVOM curve. The optimum setpoint is high enough to allow sufficient time for reliable oscillation detection, but low enough to preclude the violation of the MCPR safety limit. The setpoint determination is cycle-specific.

The licensee relies on the RAMONA5-FA computer code to calculate the CPR response of the core to regional oscillations on a cycle-specific basis. Although the RAMONA5-FA method is under generic review at the NRC, the limited application RAMONA5-FA code system for DIVOM analysis has been accepted by the staff on a plant specific basis. The licensee confirmed that the application of RAMONA5-FA for BSEP is consistent with previous applications of the code. During the audit at AREVA, the staff reviewed the DIVOM calculations performed for BSEP using RAMONA5-FA and found the results to be reasonable. AREVA's DIVOM approach

[[ ]]

]] to determine the licensing CPR response. The NRC staff finds that the stability based OLMCPR is determined for BSEP consistent with the approved NEDO-32465-A methodology.

In addition, the staff evaluated the impact of bypass voiding on the OPRM system for BSEP. The primary effect of voiding in the bypass region on the neutron detectors (e.g. LPRMs) is to reduce the detector response. High bypass voids can potentially reduce the OPRM reading, and therefore the margin to scram may increase. This may be nonconservative for stability mitigation since it may require higher amplitude oscillations to initiate an OPRM scram. In Reference 36, the licensee responded to staff RAI requesting an evaluation of bypass voiding effect on the OPRM system for BSEP.

In Reference 36, the licensee concluded that steady state and dynamic effects of bypass boiling on lowering the sensitivity of individual LPRM detectors cause

[[ ]]

]] OPRM signals used for comparison with the OPRM amplitude setpoint. The licensee performed an analysis based on the

[[ ]]

]] The staff found that these did not involve significant modeling revisions. The effect of bypass voiding on the reduction of the LPRM detector response was conservatively taken as

]]. Based on the review of the additional analysis provided by the licensee, the NRC staff has reasonable assurance that the impact of bypass voiding on the OPRM system is [[ ]].

When the OPRM system is inoperable, the plant may use an alternate approach to address stability. The current practice with the Option-III system is to use the BSP as the backup method. The BSPs include specific requirements for operator action as well as restrictions on operation in certain regions of the power/flow map. These BSP regions are determined using the STAIF methodology documented and approved by NRC in References 27 and 28. The licensee has determined the BSP regions for the upcoming Unit 1 Cycle 17, and will validate the regions on a cycle-specific basis and expand them as necessary. The NRC staff finds that the BSP approach is acceptable for BSEP.

Based on the information discussed above, the NRC staff finds that the stability analysis and evaluation performed in support of the LAR provides reasonable assurance that the proposed transition in fuel and methods will not adversely impact BSEP ability to satisfy GDC 10 and 12.

### 3.3.3 Emergency Core Cooling System (ECCS) LOCA Performance

The ECCS is designed to mitigate postulated LOCAs due to ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

The analysis methodology used for BSEP LOCA analysis is the EXEM BWR-2000 ECCS evaluation model (Reference 29), which has been approved by the NRC. The EXEM BWR-2000 methodology employs three major computer codes, RELAX, HUXY, and RODEX2, to evaluate the thermal hydraulics and fuel response during all phases of a LOCA event. RELAX (Reference 29) is used to calculate the core average channel and hot channel system response during the blowdown, refill, and reflood phases. HUXY (Reference 30) is used to perform fuel rod heatup calculations and local clad oxidation calculations at the axial plane of interest. RODEX2 (Reference 32) is used to determine fuel parameters (such as stored energy) for input to RELAX and HUXY. A complete analysis for a LOCA event starts from the specification of fuel parameters using RODEX2. RODEX2 then determines the initial stored energy for both the hot channel blowdown analysis (RELAX) and the heatup analysis (HUXY). The licensee ensures that the initial stored energy input to RELAX and HUXY is the same as or higher than that calculated by RODEX2 for the specified power, exposure, and fuel design. The thermal hydraulic response obtained from the RELAX hot channel calculation provides the boundary condition for HUXY to calculate peak cladding temperature and metal-water reaction.

In support of the LAR, the licensee performed the break spectrum analyses (Reference 10) for a core composed entirely of ATRIUM-10 fuel at beginning-of-cycle (BOC) conditions. The calculations assumed an initial core power of 102 percent of the rated power, 2923 megawatts thermal (MWt), providing a licensing basis power of 2981 MWt. The 2-percent increase reflects the maximum uncertainty in monitoring reactor power, as required by 10 CFR 50, Appendix K. The limiting fuel assembly in the core was assumed to be operated at a MAPLHGR limit of 12.5 kilowatts per foot (kW/ft). The analyses assumed an ATRIUM-10 neutronic design that is expected to be conservative, and is confirmed with each actual cycle-specific design. At full rated power, the boundaries of the MELLLA and increased core flow domain are 76.2 million pounds (mass) per hour (Mlbm/hr) and 80.5 Mlbm/hr. The analyses were performed [[ ]]

]] Therefore, the results from the analyses [[  
]] support operation within the current licensed operating domain.

The licensee stated that the break spectrum analyses presented in Reference 10 are applicable for [[

In Reference 10, the licensee stated that the [[

]] The NRC staff finds that it is reasonable to conclude that the LOCA break spectrum analysis for ATRIUM-10 fuel remains applicable through the transition cycles.

The licensee has performed a complete spectrum analysis of break sizes (1.0 Double Ended Guillotine (DEG) to 0.05 ft<sup>2</sup> split), break locations (recirculation and non-recirculation pipes), different single failures (single failures of the battery, low pressure core injection (LPCI), diesel generator, and high pressure core injection) and axial power profiles (top-peaked and mid-peaked) to maximize the peak clad temperature (PCT).

The results show the following limiting break characteristics:

- Break size/geometry: DEG / 0.8 discharge coefficient
- Break location: recirculation discharge pipe
- Single failure: LPCI
- Axial power shape: top-peaked

The PCT obtained at the above-mentioned limiting characteristics was 1923 °F.

For single recirculation loop operation (SLO), a multiplier is applied to the two-loop operation MAPLHGR limits. The application of the appropriate MAPLHGR is to assure the expected SLO PCT is less than the calculated PCT for two-loop operation. The SLO analyses are performed at BOC fuel conditions with a 0.85 multiplier applied to the two-loop MAPLHGR limit, resulting in a SLO MAPLHGR limit of 10.625 kW/ft. The limiting SLO LOCA is the 1.0 DEG recirculation discharge break with single failure of LPCI and top-peaked axial power shape. The PCT for this case is 1865 °F (less than 1923 °F) and the results are acceptable.

Based on the limiting break characteristics, the applicable ATRIUM-10 MAPLHGR limits are generated. The licensee provided the ATRIUM-10 MAPLHGR report in Reference 11. The licensee stated that the existing GE14 MAPLHGR limits are applicable for the transition cycles. The NRC staff finds this acceptable because [[

]] and the licensee confirmed (Reference 6) that no changes have been made at BSEP that would invalidate the reactor system response assumed in the GE14 MAPLHGR analysis of record.

For BSEP operation at or below the MAPLHGR limits reported in Reference 11, the ECCS performance can be summarized as follows:

- PCT: 1900 °F < 2200 °F
- Cladding oxidation: 1.16 percent < 17 percent
- Hydrogen generation: 0.5 percent < 1 percent
- Coolable geometry: maintained when all the above criteria are met
- Long-term cooling: demonstrated when the core remains flooded to the jet pump top elevation and when a core spray system is in operation.

The licensee performed BSEP specific LOCA analysis based on an NRC-approved methodology. The initial conditions, break spectrum, and power profiles selected for LOCA analysis are consistent with the NRC-approved TR, which covers sufficiently limiting scenarios to reach a maximum PCT. The NRC staff finds that the 10 CFR 50.46 acceptance criteria are met and the ECCS performance is acceptable. Based on above, the staff finds the LOCA analyses performed in support of the LAR acceptable.

#### 3.3.4 ATWS

ATWS is defined as an AOO followed by the failure of the RPS of the protection system required by GDC-20. The NRC staff reviewed the licensee's ATWS analysis to ensure that (1) the peak vessel bottom pressure is less than the ASME service level C limit of 1500 psig; (2) the peak clad temperature is within the 10 CFR 50.46 limit of 2200°F; (3) the peak suppression pool temperature is less than the design limit (220 °F for BSEP); and (4) the peak containment pressure is less than the containment design pressure (62 psig for BSEP). AREVA does not have a generically approved long-term ATWS containment evaluation methodology. Therefore, the NRC staff reviewed the licensee's long term evaluation on a cycle-specific basis.

The licensee provided plant- and cycle-specific ATWS analyses for Unit 1 Cycle 17 conditions. These analyses show acceptable results. For Unit 1 Cycle 17, the ATWS overpressure analysis was performed based on approved COTRANSA2 methodology documented in Reference 20. The licensee reported a calculated peak vessel bottom pressure for Unit 1 Cycle 17 of 1465 psig, assuming one safety relief valve was out of service. The licensing acceptance criterion of 1500 psig is met; therefore, the NRC finds it acceptable. Additionally, the licensee stated in Reference 15 that the PCT and cladding oxidation responses during an ATWS are bounded by LOCA. Based on fuel performance analyses conducted for BWR ATWS events, the NRC staff finds that there is reasonable assurance that 10 CFR 50.46 criteria will not be challenged during ATWS, and therefore finds the licensee's conclusion acceptable.

In addition to the short-term vessel overpressure and PCT analysis, the long-term suppression pool performance must be evaluated for acceptability during ATWS. Fuel design differences may impact the power and pressure excursion experienced during the ATWS event. This, in turn, may impact the amount of steam discharged to the suppression pool and containment. The licensee stated that [[

]]

The licensee provided an evaluation to compare [[

]]

The results showed that [[

]] The licensee concluded that the

fuel design difference [[

]] The licensee

therefore concluded that the introduction of ATRIUM-10 fuel will not significantly impact the long term ATWS response (suppression pool temperature and containment pressure) and the current analysis remains applicable.

The NRC staff reviewed the results of the licensee's ATWS analysis of record performed by GE. For BSEP, the limiting ATWS suppression pool temperature is 195.5°F with the design limit at 220°F. The limiting ATWS containment pressure is 12.9 psig with the design limit at 62 psig. The staff agrees with the licensee's statement that, without any system modifications, the primary affect of fuel design [[

]] the NRC staff finds that relatively minor variations in

[[ can be accommodated by the current margins available to the suppression pool and containment limits.

The NRC staff concludes that the licensee has demonstrated that the required systems are installed at BSEP and that they will continue to meet the requirements of 10 CFR 50.62. In addition, the NRC staff reviewed the information submitted by the licensee related to ATWS and concludes that the licensee adequately accounted for the effects of the proposed fuel and methodology transition on ATWS. Therefore, the NRC staff finds the proposed LAR acceptable with respect to ATWS.

### 3.4 Technical Specification Changes

#### 3.4.1 TS 1.1, "Definitions"

The proposed change involves an addition to TS Section 1.1 to provide a new definition for LHGR. The change reflects the use of AREVA fuel and methodologies. Therefore, the NRC staff finds the proposed change acceptable.

#### 3.4.2 TS 3.2.3, "Linear Heat Generation Rate (LHGR)"

The proposed change involves the addition of TS 3.2.3 to include LHGR as a required power distribution limit. The change reflects the use of AREVA fuel and methodologies. Therefore, the NRC staff finds the proposed change acceptable.

#### 3.4.3 TS 3.4.1, "Recirculation Loops Operating"

The proposed change involves an addition to TS 3.4.1 to incorporate a restriction on LHGR when in SLO. The change reflects the results of the safety analysis discussed above. Therefore, the NRC staff finds the proposed change acceptable.



#### 3.4.4 TS 3.7.6, "Main Turbine Bypass System"

The proposed change involves an addition to TS 3.7.6 to incorporate a restriction on LHGR when the turbine bypass system is inoperable. The change reflects the results of the safety analysis discussed above. Therefore, the NRC staff finds the proposed change acceptable.

#### 3.4.5 TS 5.6.5a, "Core Operating Limits Report (COLR)"

The proposed change involves an addition of "The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.3," as a required core operating limit to be documented in the cycle-specific COLR. The change reflects the use of AREVA fuel and methodologies. Therefore, the NRC staff finds the proposed change acceptable.

#### 3.4.6 TS 5.6.5b, "Core Operating Limits Report (COLR)"

The proposed change involves updating the list of approved methodologies in TS 5.6.5b to include the AREVA NP analytical methods for generating the COLR. The LAR proposes to add the following TRs to the reference list:

- XN-NF-81-58(P)(A), RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model.
- XN-NF-85-67(P)(A), Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel.
- EMF-85-74(P) Supplement 1(P)(A) and Supplement 2(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model.
- ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs.
- XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis.
- XN-NF-80-19(P)(A) Volume 4, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads.
- EMF-2158(P)(A), Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2.
- XN-NF-80-19(P)(A) Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.
- XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.
- ANF-524(P)(A), ANF Critical Power Methodology for Boiling Water Reactors.
- ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.
- ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors
- EMF-2209(P)(A), SPCB Critical Power Correlation.
- EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel.
- EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.
- EMF-2292(P)(A), ATRIUM-10: Appendix K Spray Heat Transfer Coefficients.
- EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2.
- NEDO-32465-A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications.

As discussed in Sections 3.1, 3.2, and 3.3 of this SE, the staff finds that the AREVA methodologies documented in the referenced TRs are acceptable for use in support of BSEP licensing applications. NEDO-32465-A is a BWROG TR that has been approved by the NRC staff and will continue to be used by AREVA as part of their detection and suppression analyses. Therefore, the staff concludes that the proposed list of TRs is acceptable.

#### 3.4.7 Administrative Changes

The addition of the above list of TRs added a page to TS 5.6.5. Therefore, the licensee elected to renumber current TS pages 5.0-21 to 5.0-24. This is an administrative change and is acceptable.

### 4.0 SUMMARY

The NRC staff has reviewed the LAR (Reference 1), in conjunction with the supplemental information (References 2 through 5 and 35) and the responses to the staff's RAIs (References 6, 7, and 36), to evaluate the acceptability of the BSEP transition to AREVA ATRIUM-10 fuel and AREVA safety analysis and core design methodologies. Based on its review, the NRC staff has determined that the licensee provided adequate technical basis to support the proposed TS changes. Specifically, the NRC staff finds the licensee has demonstrated that (1) BSEP complies with the staff limitations and conditions imposed for application of the TRs, (2) AREVA codes and methods are applicable for BSEP at EPU conditions, (3) the BSEP-specific safety analysis results based on the AREVA methodology meet the applicable licensing criteria, and (4) the proposed TS changes are acceptable.

### 5.0 STATE CONSULTATION

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

### 6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change the Surveillance Requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (72 FR 68208). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

### 7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the

Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

## 8.0 REFERENCES

1. Letter from J. Sarcola (Carolina Power & Light) to NRC, "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 Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA NP Fuel," January 22, 2007.
2. Letter from J. Sarcola (Carolina Power & Light) to NRC, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Docket Nos. 50-325 and 50-324/License Nos. DPR-71 and DPR-62 Additional Information in Support of Request for License Amendments Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA NP Fuel," Jun 21, 2007.
3. Letter from B. Waldrep (Carolina Power & Light) to NRC, "Additional Information in Support of Request for License Amendment Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA NP Fuel," July 18, 2007.
4. Letter from J. Sarcola (Carolina Power & Light) to NRC, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Docket Nos. 50-325 and 50-324/License Nos. DPR-71 and DPR-62 Additional Information in Support of Request for License Amendment Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA NP Fuel (NRC TAC Nos. MD4061 and MD4062)," July 31, 2007.
5. Letter from J. Sarcola (Carolina Power & Light) to NRC, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Docket Nos. 50-325 and 50-324/License Nos. DPR-71 and DPR-62 Additional Information in Support of Request for License Amendments Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA NP Fuel (NRC TAC Nos. MD4063 and MD4064)," October 15, 2007.
6. Letter from B. Waldrep (Carolina Power & Light) to NRC, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Docket Nos. 50-325 and 50-324/License Nos. DPR-71 and DPR-62 Additional Information in Support of Request for License Amendments Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA Fuel (NRC TAC Nos. MD4063 and MD4064)," January 24, 2008.
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8. AREVA NP, Inc. Topical Report ANP-2637, Revision 1, "Boiling Water Reactor Licensing Methodology Compendium," June 2007.
9. AREVA Report ANP-2646, Revision 0, "Brunswick Unit 1 Thermal-Hydraulic Design Report for ATRIUM-10 Fuel Assemblies," June 2007.

10. AREVA Report ANP-2625, Revision 0, "Brunswick Units 1 and 2 LOCA Break Spectrum Analysis for ATRIUM-10 Fuel," June 2007.
11. AREVA Report ANP-2624, Revision 0, "Brunswick Units 1 and 2 LOCA-ECCS Analysis MAPLHGR Limit for ATRIUM-10 Fuel," June 2007.
12. AREVA Report ANP-2638, Revision 0, "Applicability of AREVA NP BWR Methods to Extended Power Uprate Conditions," July 2007.
13. AREVA Report ANP-2658 Revision 0, "Brunswick Unit 1 Cycle 17 Fuel Cycle Design," July 2007.
14. AREVA Presentation Slides for Brunswick Fuel Transition License Amendment Request, August 2007.
15. AREVA Report ANP-2674, Revision 0, "Brunswick Unit 1 Cycle 17 Reload Safety Analysis," September 2007.
16. XN-NF-84-105(P)(A), Volume 1 and Volume 1, Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987.
17. XN-NF-80-19(P)(A), Volume 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, January 1987.
18. EMF-2209(P)(A) Revision 2, "SPCB Critical Power Correlation," Framatome ANP, September 2003.
19. EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000.
20. ANF-913(P)(A), Volume 1, Revision 1, and Volume 1, Supplements 2, 3 and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.
21. EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," Siemens Power Corporation, October 1999.
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23. ANF-1358(P)(A), Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, September 2005.
24. ANF-524(P)(A), Revision 2 and Supplements 1 and 2, "ANF Critical Power Methodology for Boiling Water Reactors," Advanced Nuclear Fuels Corporation, November 1990.

25. NEDO-32465-A, "Reactor Stability Detector and Suppress Solutions Licensing Basis Methodology for Reload Applications," General Electric Nuclear Energy, August 1996.
26. BAW-10255(P), Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," January 2006.
27. EMF-CC-074(P)(A), Volume 1, "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain," and Volume 2, "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain - Code Qualification Report," Siemens Power Corporation, July 1994.
28. EMF-CC-074(P)(A), Volume 4, Revision 0, "BWR Stability Analysis – Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation, August 2000.
29. EMF-2361(P)(A), Revision 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP, May 2001.
30. XN-CC-33(A), Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual," Exxon Nuclear Company, November 1975.
31. XN-NF-82-07(P)(A), Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, November 1982.
32. XN-NF-81-58(P)(A), Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984.
33. ANF-89-98(P)(A) Revision 1 and Supplement 1, Generic Mechanical Design Criteria for BWR Fuel Designs, Advanced Nuclear Fuels Corporation, May 1995.
34. XN-NF-80-19(P)(A) Volume 4 Revision 1, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, Exxon Nuclear Company, June 1986.
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36. Letter from B. Waldrep (Carolina Power & Light) to NRC, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Docket Nos. 50-325 and 50-324/License Nos. DPR-71 and DPR-62 Additional Information in Support of Request for License Amendments Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA NP Fuel (NRC TAC Nos. MD4063 and MD4064)," March 21, 2008.

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Date: March 27, 2008