Regulatory Basis:

The following Title 10 of the U.S. *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC) are applicable:

- GDC 1, "*Quality standards and records*" requires, in part, that the structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed and that there is adequate assurance that a SSC performs its safety function.

- GDC 4, "*Environmental and dynamic effects design bases,*" requires, in part, that structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

- GDC 10, "*Reactor design*," requires that: The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

- GDC 12, "*Suppression of reactor power oscillations*," requires that: The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

- GDC 28 "*Reactivity limits*," requires the reactivity control system 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 (a) result in damage to the reactor coolant pressure boundary greater than the limited local yielding, or (b) sufficiently impair core cooling capability.

SRXB-RAI-1

Please provide the BSEP-specific noise data and discussion of the licensee's evaluation of these data used to justify a Detect and Suppress Solution Confirmation Density (DSS-CD) amplitude discriminator setpoint (S_{AD}) value of **[ ]** to minimize the likelihood of spurious scram while ensuring that power oscillations can be readily detected and suppressed which, in part, ensures that fuel design limits are not exceeded.

SRXB-RAI-2

Please provide a basis for the use of a DSS-CD time period lower limit (T_{min}) of **[ ]**, which is higher than the approved value of **[ ]** given in the DSS-CD license topical report (LTR) and reduces the range of oscillations indicative of an anticipated reactor instability. This basis will ensure that power oscillations can be readily detected and suppressed which, in part, ensures that fuel design limits are not exceeded. Please include the following components:

a. Provide justification, based on plant data, that indicates that the **[ ]** value may lead to an unacceptable or undesired likelihood of spurious scram in BSEP during Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operation; and

b. Demonstrate that thermal-hydraulic (T-H) instabilities are not expected to occur below a T_{min} of [ ]. As part of this demonstration, please include analysis based on the TRACG DSS-CD analyses that were provided in the Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus (M+SAR, Enclosure 6 of the LAR), as well as a justification that the T-H oscillation period would remain above [ ] in both units, at any cycle exposure, and at any other point in the DSS-CD armed region.

SRXB -RAI-3

In the SAR, Anticipated Transient without Scram (ATWS) and ATWS Instability (ATWS-I) results were presented using the limiting fuel parameter sensitivity values to ensure that the ATRIUM 10XM fuel in BSEP was adequately modeled using GEH methods. Please justify the acceptability of [ ], for the DSS-CD confirmatory analyses. This will ensure that power oscillations can be readily detected and suppressed which, in part, ensures that fuel design limits are not exceeded for the ATRIUM 10XM fuel.

Please justify that the [ ] used in the DSS-CD approach conservatively bounds the uncertainty associated with ATRIUM 10XM fuel when the limiting fuel parameter values are considered.

SRXB-RAI-4

The following additional information is requested for the turbine trip with bypass (TTWBP) ATWS-I analyses to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the event:

a. Please describe the approach used to ensure that the maximum steady-state Linear Heat Generation Rate (LHGR) for the TRACG TTWBP analyses was less than [ ] of the Maximum LHGR limit. Were the steady state conditions (e.g. control rod positions) used to initiate the TRACG TTWBP analyses consistent with the equilibrium cycle conditions provided by AREVA?

b. What value of 'dtmax' (maximum allowed timestep size) was used in the TRACG TTWBP analyses presented in the SAR? Was this 'dtmax' value conservative for ATWS-I analyses? If not, please provide revised TRACG TTWBP analyses with an appropriate value for 'dtmax' to replace the analyses presented in the SAR.

SRXB-RAI-5

The following additional information is requested to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event:

a. Please provide a diagram of the TRACG channel grouping used for the regional and core-wide mode TTWBP ATWS-I analyses. Did the regional mode analyses result in higher PCT values than the core-wide analyses?

b. Figure 9-10 of the SAR provides total core values for neutron flux and inlet flow rate versus time; however, this does not sufficiently describe the local assembly behavior during regional mode oscillations. Please provide additional time-dependent results at symmetric core locations (including the limiting Peak Cladding Temperature (PCT) assembly) showing the amplitude of the regional oscillations for the revised TTWBP analyses requested in SRXB-RAI-4. Please include time-dependent assembly power,

assembly inlet flow rate, and maximum cladding temperature for each of these two assemblies, as well as the axial location where the maximum PCT occurs.

SRXB-RAI-6

Please provide the following information regarding the limiting fuel parameter sensitivities to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event:

a. Please provide the process used to determine the parameters used for the ATRIUM 10XM fuel parameter sensitivity study for the ATWS and ATWS-I analyses for BSEP MELLLA+. For each parameter, please provide information on how the sensitivity range was determined.

b. Please provide a table showing the maximum PCT value for the TRACG TTWBP analyses (including the revisions described in SRXB-RAI-4, if necessary) obtained by adjusting each fuel parameter individually to the minimum and maximum value within the appropriate sensitivity range to demonstrate each parameter's impact on the results.

SRXB-RAI-7

Based on recent NRC funded ATWS-I test experiments (KATHY), the failure to rewet (FTR) temperature is in reasonable agreement with homogeneous nucleation plus contact temperature and provides a reasonable representation of the cladding temperature behavior during ATWS-I oscillations. The homogeneous nucleation plus contact temperature is more conservative compared to Modified Shumway T_{min} model used in the SAR. Therefore, the following sensitivity studies are requested to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the event

a. Please provide TRACG sensitivity studies for the TTWBP event in BSEP using the homogeneous nucleation temperature plus contact temperature model for T_{min}. Sensitivity studies may include realistic assumptions for input parameters such as feedwater temperature versus time, operator action time to reduce water level, maximum initial LHGR value, and use of the TRACG quench model. Please also provide sensitivity studies using the limiting fuel parameter sensitivity values in conjunction with the homogeneous nucleation T_{min} model.

b. In the following paragraph, "limiting fuel parameter delta-PCT" refers to the PCT when using the limiting fuel parameter values minus the PCT when using nominal fuel parameter values. If the limiting fuel parameter delta-PCT in any of the cases performed in (a) was larger than the limiting fuel parameter delta-PCT reported in the SAR, please explain why the limiting fuel parameter delta-PCT was larger. This explanation may include consideration of the location of maximum PCT, the average or peak LHGR at this location, or other considerations as appropriate.

SRXB-RAI-8

Please justify the feedwater temperature reduction rate that was used in the ATWS-I TTWBP analyses presented in the SAR, as well any reduced rate used in the ATWS-I sensitivity studies to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event. Justify this using appropriate plant data or simulator results, if available. If plant simulator data is provided, please justify the adequacy of the simulator for determining the feedwater temperature reduction rate.

SRXB-RAI-9

To ensure that the event is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event, the staff has the following questions regarding the use of the GEXL correlation for ATWS-I analyses:

a. Please discuss the GEXL correlation's applicability to oscillatory conditions.

b. For ATWS-I oscillations such as those calculated in the TTWBP analyses for BSEP MELLLA+, is the critical heat flux temperature (in the oscillation growth phase and in the limit cycle phase) during a given oscillation period typically determined by **[ ]**

SRXB-RAI-10

To ensure that the event is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event, the staff requests the following additional information regarding the use of R-factors in the TRACG ATWS-I analyses:

a. Please provide the R-factors used in each assembly in the TRACG TTWBP models, and justify how these R-factors were determined. If applicable, please provide TTWBP sensitivity results that indicate the effect that changing the GEXL critical power value has on maximum PCT during this event.

b. The stated range of validity of R-factors in the GEXL97 correlation for ATRIUM-10 fuel is **[ ]**. If an R-factor less than **[ ]** was used in the ATWS-I analyses, please justify the use of these ATRIUM 10XM R-factors outside of the stated range of validity. Does the GEXL correlation behave properly for R < **[ ]**? (For example, is the **[ ]** term intended to be used for R < **[ ]**?)

c. For safety analyses such as anticipated operational occurrences (AOOs), higher hot-rod R-factor values are typically more limiting. If a lower R-factor value was found to be more limiting in Part (a), please explain why it was more limiting or please provide additional sensitivity analyses using the following assumptions for assembly R-factors, to better understand the limiting combination of R-factors for the TTWBP event:

    1. Hot channel R-factor taken from bounding high value from AREVA inputs, and remaining R-factors taken as the low value

    2. Hot channel R-factor taken as the low value, and remaining R-factors taken as the bounding high value

    3. All R-factors taken as the low value

    4. All R-factors taken as the bounding high value

SRXB-RAI-11

Please provide steady-state core simulator comparisons for a representative BSEP MELLLA+ cycle using GEH and AREVA methods, to support the licensee's conclusion that the GEH methods have modeled ATRIUM 10XM in a satisfactory manner. Include comparisons of calculated results such as power and exposure distributions, hot eigenvalue, active or bypass flow rates, and core pressure drop.

SRXB-RAI-12

Please provide justification that safety limit minimum critical power ratio (SLMCPR) Penalty of 0.03 is not applicable to BSEP. Please provide data representative of the requested MELLLA+

Operating Domain for BSEP to justify the use of MICROBURN-B2 in the domain. If the data doesn't cover the entire operating domain, please identify the range that is not covered and identify any additional conservatisms that can be used to justify the adequacy of the method in this range.

SRXB-RAI-13

AREVA ANP-3280P - *Brunswick Unit 1 Cycle 19 MELLLA+ Reload Safety Analysis, May 2016*, includes reference reload analyses for a mixed ATRIUM 10XM/ATRIUM-10 core. However, Section 1.0 of ANP-3280P indicates that ATRIUM-10 limits and analyses are not included in these reload analyses. Please explain this approach in more detail to ensure that it is appropriate for a full core of ATRIUM 10XM which will be the fuel type when MELLLA+ is implemented. For example, were the geometric and performance features of the ATRIUM-10 fuel assemblies modeled explicitly, while the SLMCPR calculations (such as those shown in Table 4.2 of ANP-3280P) considered only the ATRIUM 10XM fuel assemblies for Cycle 18? How does Section A.1 of AREVA ANP-3108P *Applicability of AREVA NP BWR Methods to Brunswick Extended Power Flow Operating Domain, July 2015*, regarding mixed cores relate to the analyses performed in ANP-3280P?

SNPB-RAI-1

Section 6.0 of ANP-3108 *Mechanical Limits Methodology* describes how the fuel mechanical design criteria are satisfied. Provide the details for the following aspects of the mechanical design for the extended power/flow operation at the Brunswick units:

a. Provide a summary description how fuel rod design criteria was applied for Extended Power/Flow Operating Domain (EPFOD) operation.

b. How the fuel design limits such as LHGR and burnup are established for the EPFOD operation.

c. Provide details of how the uncertainties (operating power, code model parameters, and fuel manufacturing tolerances) are utilized in the mechanical design analysis. Also, list the values and source of uncertainties utilized in the analysis.

SNPB-RAI-2

Background

Section 6.2 of ANP-3280P, Revision 1, "Brunswick Unit 1 Cycle 19 MELLLA+ Reload Safety Analysis," provides a short summary of Control Rod Drop Accident (CRDA). The CRDA analysis for both A and B sequence startups was performed/dispositioned using the methodology described in Topical Report (TR), XN-NF-80-19(P)(A) Volume 1 Supplements 1 & 2, "Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis", (1983). The licensee has reported that that the maximum fuel rod enthalpy is less than the NRC threshold of 280 cal/g.

Appendix B of Standard Review Plan (NUREG-0800, March 2007) provides interim acceptance criteria for the reactivity initiated accidents, such as, CRDA for BWRs. The technical and regulatory basis for the interim criteria is documented in a memorandum dated January 19,

2007 (ADAMS Number ML070220400). This memorandum, with respect to the criteria on fuel enthalpy for maintaining coolable geometry, states that the 280 cal/g criteria has been found inadequate to ensure fuel rod geometry and long term coolability. NUREG-0800 Section 4.2 defines reactivity-initiated accident (RIA) fuel clad failure criteria as (1) radial average fuel enthalpy greater than 170 cal/g for BWR at zero or low power, and (2) local heat flux exceeding fuel thermal design limits (CPR) at-power events in BWRs. The technical basis for the BWR fuel failure criteria is detailed in the January 19, 2007 memorandum.

The draft regulatory guide (DG-1327 of November 2016) defines fuel cladding failure thresholds, analytical limits and guidance for demonstrating compliance with applicable regulations governing reactivity limits. The empirically based pellet-cladding mechanical interaction (PCMI) failure thresholds are shown in Figures 2 through 5 for fully recrystallized annealed (RXA) and stress relief annealed (SRA) cladding types at both low and high temperature reactor coolant conditions. The PCMI cladding failure threshold is expressed in peak radial average fuel enthalpy rise (Δcal/g) versus excess cladding hydrogen content in weight parts per million (wppm).

Based on the background information given, please provide an evaluation to show that the fuel failure criteria as proposed by either SRP Section 4.2 (2007) or DG-1327 (2016) can be met for the control rod drop accident analysis at BSEP Units 1 and 2.

SNPB-RAI-3

Appendix B of ANP-3108P describes void-quality correlations, **[** **]** and Ohkawa-Lahey, and AREVA's independent validation of these correlations for application to ATRIUM 10XM and Brunswick plant operation at EPFOD conditions. Please provide responses for the following items.

a. Detailed description of AREVA's independent validation of the **[** **]** and Ohkawa-Lahey correlations using the test data from FRIGG experiments that generated Table B-1 and Figures B-1 through B-4 of ANP-3108P.

b. Supporting calculation detail that shows how the uncertainties and biases are utilized in the analyses described in Sections B.2 and B.3 of ANP-3108P.

SNPB-RAI-4

It has been stated in Section 9.3.3 *ATWS with Core Instability* of DUKE-0B21-1104-000(P) (M+ LTR) that sensitivity studies are performed with GEXL by varying from **[** **]**, and conservative GEXL parameters have been developed by AREVA for Application to ATRIUM 10XM fuel. The staff could not find any reference to this GEXL correlation and could not determine which version of GEXL correlation has been utilized and which sensitivity analysis has been performed.

a. Please provide details of the formulation of the above GEXL correlation formulation.

b. Do the biases and uncertainties associated with this GEXL correlation comply with the requirements for the ATRIUM 10XM fuel design?

c. Is this GEXL correlation compatible with TRACG code?

d. Are the R-factors (K-factors) and additive constants specifically derived for the ATRIUM 10XM fuel design?

SNPB-RAI-5

Appendix A page A-2 indicates that "*At the time of the creation of this document, Reference 7 (ANP_10298PA Revision 0 Supplement 1P Revision 0 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, December 2011) had not been generically approved.*" The staff notices that Revision 1 of ANP-10298P-A was published as the accepted version (-A) incorporating Reference 7 in to Reference 2 (ANP-10298PA Revision 0) in March 2014. However, the Brunswick EPFOD analysis was performed using the Revision 0 when Revision 1 of the topical report was available. Please explain why the Revision 1 was not used for the Brunswick analysis. What is the impact on results if the Revision 1 is used in the analysis?

SNPB-RAI-7

For ATWS (Licensing basis) calculations (Section 9.3.1 of DUKE-0B21-1104-000(P) (M+ LTR)) and for ATWS with core instability (Section 9.3.3 of M+LTR), to address the effect of ATRIUM 10XM fuel, sensitivity analyses were performed. Sensitivity ranges are selected to include expected variation of the parameters; direct energy deposition, gap conductance as applied to ATRIUM 10XM as supplied by PRIME, thermal and hydraulic channel losses. Please respond to the following questions.

a. Discuss the suitability of using ODYN code for ATRIUM 10XM fuel design.

b. It is stated that gap conductance data as applied to ATRIUM 10XM fuel data is supplied by the PRIME code. However, Section 4.0 of safety evaluation for PRIME *limits the use of PRIME to "approved GNF fuel rods designs clad in RXA Zircaloy-2."* This limitation further states that *"In case core transition from one vendor to another, PRIME may be applied to generate inputs for the downstream safety analyses and overpower limit compliance for the non-GNF BWR fuel, provided that the design and operating parameters for the non-GNF fuel must be within the range approved for the PRIME models;*
   1. Since the cladding type for the ATRIUM 10XM is different from the cladding type of GNF fuel designs (RXA), please justify how PRIME can be used for the ATRIUM 10XM analysis.
   2. To satisfy the limitation for PRIME stated above, show that the ATRIUM 10XM design and operating parameters are within the ranges of the parameters approved for PRIME

In order to use PRIME for the ATRIUM 10XM analyses, has any modifications been performed in the PRIME code to enable it to perform ATRIUM 10XM analyses?

EMIB-RAI-1

AS per the LTR, NEDC-33006P-A, Rev.3, The relief and safety valves and the reactor protection system provide overpressure protection for the reactor coolant pressure boundary (RCPB) during power operation. The NRC staff's review covered relief and safety valves on the

main steam lines and piping from these valves to the suppression pool. The NRC's acceptance criteria are based on (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (2) draft GDC-33, -34, and -35, insofar as they require that the RCPB be designed to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating type failures is minimized. Specific review criteria are contained in SRP Section 5.2.2 and other guidance provided in Matrix 8 Table of RS-001, "Review Standard for Extended Power Uprates," (The pressure relief systems provide reactor overpressure protection for the NSSS to prevent failure of the nuclear system pressure boundary and uncontrolled release of fission products, during abnormal operational transients, the ASME Upset overpressure protection event, and postulated ATWS events. Section 9.3.1 of the M+SAR evaluates the ATWS response for operation at the MELLLA+ operating domain.

Since the evaluation of the steam separator and dryer performance at MELLLA+ conditions indicates an increase in moisture carry over (MCO) of < 0.20 wt% where the original MCO performance specification was 0.10 wt. Please discuss the impact of the higher moisture concentration on components in the main steam lines, including MSIVs and flow restrictors.