June 21, 2007

Ronnie L. Gardner, Manager
AREVA NP
3315 Old Forrest Road
P.O. Box 10935
Lynchburg, VA 24506-0935

SUBJECT: DRAFT SAFETY EVALUATION REPORT FOR TOPICAL REPORT ANP-10263(P), "CODES AND METHODS APPLICABILITY REPORT FOR THE U.S. EVOLUTIONARY POWER REACTOR (U.S. EPR)" (TAC NO. MD2803)

Dear Mr. Gardner:

By letter dated August 10, 2006 (ML062270392), as supplemented by letter dated October 6, 2006 (ML062850040), AREVA NP (AREVA) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review proprietary (ML062270395) and non-proprietary (ML062270393) versions of Topical Report (TR) ANP-10263, Revision 0, "Codes and Methods Applicability Report for the U.S. EPR." Enclosed for your review and comment is a copy of the staff's draft safety evaluation (SE) for the TR.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 2.390, we have determined that the enclosed draft SE does not contain proprietary information. However, we will delay placing the draft SE in the public document room for a period of ten working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects. If you believe that any information in the enclosure is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. After ten working days , the draft SE will be made publicly available, and an additional ten working days are provided to you to comment on any factual errors or clarity concerns contained in the SE. The final SE will be issued after making any necessary changes and will be made publicly available. The staff's disposition of your comments on the draft SE will be discussed in the final SE.

To facilitate the staff's review of your comments, please provide a marked-up copy of the draft SE showing proposed changes and provide a summary table of the proposed changes.

If you have any questions, please contact me at gxt2@nrc.gov or (301) 415-3361.

Sincerely,

/RA/

Getachew Tesfaye, Sr. Project Manager
EPR Projects Branch 1
Division of New Reactor Licensing
Office of New Reactors

Project No. 733

Enclosure: Draft Safety Evaluation

cc w/encl: U.S. EPR Service List

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SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS

TOPICAL REPORT NUMBER ANP-10263(P), REVISION 0

“CODES AND METHODS APPLICABILITY REPORT FOR THE U.S.

EVOLUTIONARY POWER REACTOR (U.S. EPR)” (TAC NO. MD2803)

PROJECT NO. 733

1.0 INTRODUCTION

By letter dated August 10, 2006 (ML062270392), as supplemented by letter dated October 6, 2006 (ML062850040), AREVA NP (AREVA) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review proprietary (ML062270395) and non-proprietary (ML062270393) versions of Topical Report (TR) ANP-10263, Revision 0, “Codes and Methods Applicability Report for the U.S. EPR.” AREVA requested that the NRC issue a safety evaluation report which approves the use of previously accepted AREVA Pressurized Water Reactor (PWR) codes and methods described in the TR, when appropriately modified for the U.S. EPR. AREVA plans to reference the approved version of the TR in its Design Control Document for the U.S. EPR.

AREVA personnel identified an error in the input deck for the small break loss-of-coolant accident (SBLOCA) sample problem presented in Appendix C of the TR. This error has been entered into AREVA's corrective action program. The evaluations and conclusions presented in the main body of the report were unaffected. AREVA submitted the revised results of the sample problem by a letter dated November 1, 2006 (ML063100173), “Revised Appendix C for Topical Report ANP-10263P Revision 0.”

The staff requested additional information (RAI) to clarify and supplement the information presented in the TR in a letter dated February 28, 2007 (ML070520239), “Request for Additional Information Regarding Topical Report ANP-10263(P), Codes and Methods Applicability Report for the U.S. Evolutionary Power Reactor (U.S. EPR).” AREVA provided responses to the RAI by letters dated March 28, 2007 (ML070920116) and April 24, 2007 (ML071160125).

ANP-10263 addresses the primary fuel codes and methods to be used for U.S. EPR analyses. The fuel analysis codes include:

- CASMO3 and PRISM – used for core neutronics (“Reactor Analysis System for PWRs,” EMF-96-029(P)(A), January 1997),

- NEMO-K – used for core reactivity transients (“NEMO-K, A Kinetics Solution in NEMO,” BAW-10221P-A, September 1998),

Enclosure

- LYNXT – used for detailed core thermal/hydraulic analyses ("LYNXT – Core Transient Thermal-Hydraulic Program," BAW-10156P-A Revision 1, August 1993), and

- COPERNIC – used for core thermal/mechanical analyses ("COPERNIC Fuel Rod Design Computer Code," BAW-10231P-A, June 2002).

ANP-10263 also addresses the codes and methods to be used for Appendix K-based SBLOCA analyses and for non-LOCA [loss-of-coolant accident] analyses. The large break LOCA methodology is not addressed in this TR. The S-RELAP5 code will be used for SBLOCA analyses ("PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," EMF-2328(P)(A) Revision 0, March 2001), and for non-LOCA analyses ("[Standard Review Plan] SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," EMF-2310(P)(A) Revision 1, May 2004).

2.0 REGULATORY EVALUATION

There are no specific regulatory requirements for the review of TR submittals. The staff's review was based on an evaluation of the technical merit of the material provided and compliance with any applicable regulations associated with the material presented.

The staff considered Section 4.3, "Nuclear Design " of the SRP for its review of the primary fuel codes and methodologies.

The staff has previously reviewed and accepted the CASMO3 and PRISM code, in a letter dated October 29, 1996, to Mr. H. D. Curet, Siemens Power Corporation, "Acceptance for Referencing of Topical Report EMF-96-029(P) Vols 1 and 2, Reactor Analysis System for PWR'S (TAC No. M95745)."

The staff has previously reviewed and accepted the NEMO-K code, in a letter dated June 10, 1998, to Mr. F. McPhatter, Framatome Technologies, "Acceptance for Referencing of Licensing Topical Report BAW-10221 P, NEMO-K, A Kinetics Solution in Nemo."

The staff has previously reviewed and accepted the LYNXT code, in a letter dated December 3, 1985, to Mr. J. H. Taylor, Babcock & Wilcox, "Acceptance for Referencing of Licensing Topical Report BAW-10156, 'ULYNXT- Core Transient Thermal-Hydraulic Program'." The staff also reviewed and accepted Revision 1 of the LYNXT code, Topical Report BAW-10156A, Revision 1, dated August 1993.

The staff has previously reviewed and accepted the COPERNIC code, in a letter dated June 14, 2002, to Mr. James Mallay, Framatome ANP, "Framatome ANP Topical Report BAW-10231: COPERNIC Fuel Rod Design Computer Code -Correction of Error in Safety Evaluation (TAC NO. MA6792)."

The staff did not specifically consider Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," or Appendix K to Part 50, "ECCS [emergency core cooling system] Evaluation Models," for its review of the proposed SBLOCA code or methods. The review centered on U.S. EPR design features which might require new models or methods for licensing analyses. The staff did not specifically consider Chapter 15, “Accident Analysis,” of the SRP for its review of the proposed non-LOCA code or methods. The review centered on U.S. EPR design features which might require new models or methods for licensing analyses.

The staff has previously reviewed and accepted the S-RELAP5 code for Appendix K-based SBLOCA evaluations in a letter dated March 15, 2001 (ML010800365), to Mr. James F. Mallay, Framatome ANP, “Acceptance for Referencing of Licensing Topical Report EMF-2328(P), Revision 0, PWR Small Break LOCA Evaluation Model, S-RELAP5 Based (TAC No. MA8022).”

The staff has previously reviewed and accepted the S-RELAP5 code for non-LOCA analyses in a letter dated May 19, 2004 (ML041400499), to Mr. James F. Mallay, Framatome ANP, “Final Safety Evaluation for Topical Report EMF-2310(P), Revision 1, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors (TAC No. MC0329).”

3.0 TECHNICAL EVALUATION

The U.S. EPR is an evolutionary PWR design, generally similar to existing PWRs. AREVA presented information in ANP-10263 to demonstrate the similarity of the fuel and the reactor coolant system (RCS) designs to currently operating reactors. In addition, AREVA described some differences between the U.S. EPR and existing reactor designs that are expected to enhance the safety performance of the U.S. EPR during transients and accidents.

Because of the similarity between the U.S. EPR and existing reactors, AREVA believes that the computer codes and methods approved by the NRC for the analysis of existing reactors can be justified as appropriate for the safety analyses of the U.S. EPR.

3.1 Comparison of U.S. EPR to U.S. 4-Loop PWR

Table 1 below provides a comparison of some of the design parameters for the U.S. EPR and a typical current generation U.S. 4-Loop PWR.

Table 1 - Comparison of Key U.S. EPR Design Parameters to U.S. 4-Loop PWR

Parameter	U.S. EPR	U.S. 4-Loop PWR
Design life (yrs)	60	40
Thermal power (MW(t))	4,500	3,411
Hot leg temp (°F)	624	610
Cold leg temp (°F)	564	547
RCS flow per loop (gpm)	125,000	90,000
PZR volume (ft^3)	2,650	1,800
SG secondary inventory (lbm)	182,000	106,000
Primary system pressure (psia)	2,250	2,250

Parameter	U.S. EPR	U.S. 4-Loop PWR
Steam pressure (psia)	1,118	870
Steam flow per loop (Mlbm/h)	5.1	3.8
Total RCS volume (ft^3)	16,245	12,600
Number of fuel assemblies	241	193
Fuel lattice	17 x 17	17 x 17
Active fuel length (ft)	13.78	12.00
Rods per assembly	265	264
Average linear heat rate (kW/ft)	4.98	5.44
Best estimate peak operating linear heat rate (kW/ft)	12.95 (total peaking factor 2.6)	13.06 (total peaking factor 2.4)
Number of RCCAs	89	53
Primary volume/power (ft^3/MW(t))	3.61	3.69
Secondary mass/power (lbm/MW(t)/SG)	40.4	31.1
PZR steam-to-RCS liquid volume	0.070	0.061
LOCA break area/system volume (1/ft)	3.17×10^{-4}	3.27×10^{-4}
Accumulator volume/RCS volume	0.35	0.30
Containment free volume (ft^3)	2.82×10^6	2.62×10^6

The major design differences between the U.S. EPR and the 4-loop PWR were addressed by AREVA in responses to RAIs.

<u>Core Power</u>: U.S. EPR core power is higher, however the average linear power density is lower. The higher power is achieved by increasing the number of fuel assemblies in the core and the fuel assembly length, which increases the heat transfer area. The core heat transfer area is a plant specific input to the thermal-hydraulic methodologies and does not affect the applicability or application range of the thermal-hydraulic methodologies.

<u>RCS Loop Flow</u>: Increased RCS loop flow and total core flow maintaining fluid velocities and temperatures within the core and steam generators consistent with current plant operation to support the power increase. The core flow area is increased so that the pressure drops and heat fluxes, which depend on local fluid velocities, are within the typical range of current PWRs.

<u>Steam Pressure and Flow</u>: Because of the power increase, secondary system steam flow is also higher for the U.S. EPR, and the secondary pressure is increased. The steam generator flow area is increased so that the pressure drops and heat fluxes, which depend on local fluid velocities, are within the typical range of current PWRs.

<u>RCS and Pressurizer Volumes and Inventories</u>: The parameter of interest is the ratio of component fluid volume to reactor power. The RCS coolant volume to power ratio parameter is shown in Table 1 to be within the range of current PWRs. The pressurizer and steam generator volume-to-power ratios are higher than current PWRs. The increased water inventories improve the system response to some transients.

<u>Steam Generator Secondary Design</u>: The axial economizer in the secondary side of the steam generators is a hardware change from the current PWR designs. The economizer separates the steam generator secondary into regions having somewhat different fluid temperatures during normal operation. The economizer region is defined through input to the S-RELAP5 code.

<u>Heavy Reflector Replacing the Core Barrel Thermal Shield Pads</u>: The space between the reactor core and the core barrel is filled with a heavy neutron reflector to reduce neutron leakage and flatten the power distribution. The heavy reflector replaces the thermal shield structures found in some current PWRs. The heavy reflector is a stainless steel structure made of rings piled up one on top of the other, which are keyed together and axially restrained by tie rods bolted to the core support plate. The heat generated inside the steel structure, by absorption of gamma radiation, is removed by primary coolant flow through holes and gaps in the heavy reflector structure. The S-RELAP5 model of the U.S. EPR reflects this design. The effect of the metal in the reflector will be represented by standard heat conductors. The bypass flow through the coolant channels in the reflector metal structure will be simulated by suitable standard flow paths. In their RAI responses, AREVA stated that the flow path characteristics are set to yield a core bypass steady state flow rate through the heavy reflector typical of the core bypass flow rates in current PWR designs.

In response to the staff's RAI, AREVA addressed the potential impact of the design differences on the applicability of the S-RELAP5 thermal-hydraulic code, which was developed and approved for plants similar to the 4-loop PWR. Both SBLOCA (including a steam generator tube rupture (SGTR) accident) and non-LOCA transients and accidents were considered.

Based on this evaluation, the staff finds there is reasonable assurance that the response of the U.S. EPR to SBLOCAs and non-LOCA transient and accidents can be evaluated with the currently approved S-RELAP5 code. However, the modeling of some features, notably the steam generators, is to be addressed by AREVA prior to, or during, the Design Certification review (See Sections 3.3 and 3.4 below for additional information).

AREVA concurred with the staff's observation that the writing style in the non-LOCA section of the report created the impression in some statements that AREVA had reservations. In response to RAI 9, AREVA stated that this was unintended. AREVA proposes to review the section for statements such as "would be considered" and "is expected to" and strengthen them in the approved version of the TR. The staff finds this acceptable.

3.2 Validation of Fuel Analysis Codes for U.S. EPR

Table 2 below provides a comparison of some of the core and neutronic data for the U.S. EPR and a typical current generation U.S. 4-Loop PWR.

Table 2 - Comparison of Key U.S. EPR Core and Neutronic Data to U.S. 4-Loop PWR

Core and Neutronic Data	U.S. EPR	U.S. 4-Loop PWR
Preliminary Core Parameters		
Fuel assembly pitch in core (cold) (in)	8.466	8.466
Fuel assembly length (cold, without springs) (ft)	15.76	13.24
Average linear power (at rated power) (kW/ft)	4.98	5.44
Diameter of fuel rods (in)	0.374	0.374
Number of guide tubes per assembly (—)	24	25
Guide tube outer diameter (in)	0.490	0.482
Number of spacer grids per assembly (—)	10	8
Fuel rod pitch (in)	0.496	0.496
Total fuel rod length (cold) (ft)	14.93	12.65
Total fuel assembly mass (lbm)	1,731	1,426
Assembly UO_2 mass (lbm)	1,338	1,153
Active Core		
Equivalent diameter (ft)	12.35 ft	11.06 ft
Height-to-diameter ratio (—)	1.115	1.09
Total cross-section area (ft^2)	119.96	96.04
Fuel Rods		
Number (—)	63,865	51,145
Outside diameter (in)	0.374	0.374
Diametral gap (in)	0.0065	0.0065
Clad thickness (in)	0.0225	0.0225
Clad material	M5®	Zircaloy-4
Co-Mixed Burnable Poison (Typical) Gd_2O_3		
Material	Gd_2O_3	Gd_2O_3
Gadolinium concentration (wt%)	2 – 8	2 - 8
UO_2 carrier enrichment (wt% ^{235}U)	~0.7 (host enrichment*)	~0.7 (host enrichment*)

*Host enrichment is the non-gadolinium fuel pin enrichment.

PWR fuel analyses consist of three distinct engineering disciplines: (1) neutronic analysis; (2) thermal-hydraulic analysis; and (3) thermo-mechanical analysis. Each discipline is governed by several methodology TRs, each of which employ one or more major codes or a system of codes to perform the design calculations. Each of the referenced methodology TRs has been reviewed and accepted for use by the NRC for current PWR designs.

The codes and methodologies have also been applied to a variety of commercial reactor designs for which AREVA is currently providing or has recently provided fuel. These designs encompass a wide variety of fuel and core configurations and operating parameters. The U.S. EPR design is geometrically, functionally, and phenomenologically similar to these designs, see Table 2 above. These similarities lend technical justification to extending the currently approved methodologies to the U.S. EPR. The following section addresses the neutronic codes and methods. The other fuel related analyses disciplines stated above are addressed in Sections 3.5 and 3.6 of this report.

3.2.1 Neutronics

PWR fuel neutronics analyses fall within three general analytical areas:

(1) Core design/core follow;
(2) Power distribution monitoring; and
(3) Safety analyses/thermal-hydraulic/thermo-mechanical support.

Several neutronics methodology TRs were developed by AREVA, including “Reactor Analysis System for PWRs,” EMF-96-029(P)(A), Volumes 1 and 2, January 1997, and “NEMO-K A Kinetics Solution in NEMO,” BAW-10221P-A, September 1998. The codes and methods were reviewed and accepted by the NRC to support these types of analyses. Table 3-1 in ANP-10263 provides a road map between the various neutronic design TRs intended for use in the U.S. EPR Design Certification and the analysis area(s) in which they are used.

3.2.2 Neutronics Codes

Each of the neutronic methodologies use one or more computer codes to perform the analyses. The codes have been accepted by the NRC for use as part of the review and acceptance of the TRs noted above. The major codes used are CASMO-3G/MICBURN-3, PRISM and NEMO-K. A brief description for each code is provided below.

CASMO-3G is an industry standard multi-group, two-dimensional transport theory code used for burnup calculations on PWR and boiling water reactors (BWR) fuel assemblies and/or simple pin cells. It is used in conjunction with the MICBURN-3 code which is a multi-group, one-dimensional transmission probability code that is used to calculate microscopic burnup in rods containing absorber (poison) material that is initially homogeneously distributed. The code system is designed to handle a geometry consisting of cylindrical fuel rods of varying compositions in a square pitch array with allowance for fuel rods loaded with a burnable absorber, burnable absorber rods, cluster control rods, in-core instrument channels, water gaps, boron steel curtains, and cruciform control rods in the regions separating fuel assemblies. The code system is used to generate assembly lattice cross section data as a function of burnup, discontinuity factors, and heterogeneous form factors which are then passed to the core simulator code (PRISM).

PRISM is a three-dimensional core simulator code used to calculate core power distributions and perform reactivity analyses using assembly data from the CASMO-3G code. The code uses a nodal expansion method to solve the two-group neutron diffusion theory representation of the reactor core. It provides for thermal-hydraulic feedback, modeling of equilibrium or time-dependent xenon and samarium, and isotopic depletion. In addition, it allows for the generation of pin-by-pin power distributions using a pin power reconstruction technique. A continuous cross section representation was developed covering all combinations of thermal hydraulic parameters possible in stationary reactor states. The code has been thoroughly benchmarked against startup physics test results and core follow data obtained from several domestic and foreign commercial power reactors.

NEMO-K is a three-dimensional, reactor kinetics code incorporating time-dependent solutions for neutronics, fuel temperature, and coolant properties into the steady-state NEMO computer code. NEMO-K solves the nodal balance equations in three dimensions to determine the neutron flux, source, relative power density (including pin power reconstruction), and reactor core reactivity. The code tracks both the power generated within the fuel and that deposited directly in the water. The fuel pin temperature model assumes a circular rod with azimuthally symmetric heat generation and no axial conduction. The transient thermal-hydraulic model assumes no mixing between channels and a constant pressure throughout the core.

3.3 Proposed Methodology for Appendix K Small Break LOCA

LOCAs are design basis accidents (DBAs) that would result from the loss of reactor coolant from piping breaks in the RCS pressure boundary at a rate exceeding the capacity of the normal reactor coolant makeup system. Breaks are postulated to occur at various locations and include a spectrum of break sizes. A significant loss of reactor coolant inventory would degrade heat removal from the reactor core unless the water is replenished. The NRC's General Design Criterion 35 requires each PWR be equipped with an ECCS that refills the vessel in a timely manner to satisfy the requirements of the regulations for ECCS given in 10 CFR Part 50 (Section 50.46 and Appendix K).

SBLOCA are defined as any break in the RCS with a flow area of up to 10% of a cold leg pipe area. A break in the cold leg of an RCS is considered to be most limiting because it causes the greatest direct loss of ECCS fluid through the break and core venting is delayed because of the “loop seals” in the RCS. Loop seal behavior is important for a SBLOCA DBA. The liquid level is depressed to the level of the top of the horizontal loop seal piping. Once the loop seal clears, steam previously trapped in the upper parts of the RCS can be vented to the break. The break flow transitions from a low quality mixture to primarily steam. For some break sizes, the inner vessel mixture level can fall prior to loop seal venting, causing a short period of core uncovery. Following loop seal clearing, the core mixture level recovers to the cold leg elevation, as the steam pressure is relieved.

The U.S. EPR does not have traditional high head safety injection pumps as part of the ECCS system. The U.S. EPR relies on safety-related steam generator crash cooling to bring the primary system pressure down to the shut-off head of the medium head safety injection pump. The Main Steam Relief Train is also used to manage an SGTR event. Therefore, the modeling of the steam generator, including reflux condensation and natural circulation, was the focus of the staff’s review for the SBLOCA.

The vertical and horizontal flow maps in S-RELAP5 are described in Section 3.1 of EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based." Interphase friction is described in Section 3.2 of EMF-2328, including entrainment. The models have been assessed by AREVA and have been found acceptable for SBLOCA evaluations.

Tests and studies have shown the following:

- Non-uniform flow conditions can be expected in the steam generator during reflux condensation. Some tubes may be in a condensing co-current two-phase flow pattern while others stagnate. This has been found to cause a reduction in the effective heat transfer area in the steam generator. Examples may be found in the following studies:

  "Heat Transfer Characteristics of Reflux Condensation Phenomena in a Single Tube," G-H. Chou, J-C. Chem, Nuclear Science and Engineering: 127, 220-229 (1997).

  "Non-uniform flow distribution in steam generator U-tubes of a pressurized water reactor plant during single- and two-phase natural circulation," J-J. Jeong, et al., Nuclear Science and Engineering: 231, 303-314 (2004).

  "Nonuniform Steam Generator U-Tube Flow Distribution During Natural Circulation Tests in ROSA-IV Large Scale Test Facility," Y. Kukita, et al., Nuclear Science and Engineering: 99, 289-298 (1988).

  "Intentional Depressurization of Steam generator Secondary Side during a PWR Small-Break Loss-of-Coolant Accident," H. Asaka, Y, Hukita, Journal of Nuclear Science and Technology: 32[2]. pp. 101-110 (February 1995).

  "Thermal-hydraulic characteristics of a next generation reactor relying on steam generator secondary side cooling for primary depressurization and long-term passive core cooling," Nucl. Eng. Design,185, pp. 83-96, (1998).

- Noncondensable gases can accumulate in the U-tubes. These noncondensables can originate from air initially dissolved in the primary coolant. Local condensation heat transfer coefficients are known to decrease as the noncondensable gas mass fraction increases. Examples may be found in the following studies:

  "Reflux condensation behavior in a U-tube steam generator with or without noncondensables," Tay-Jian Liu, Nuclear Engineering and Design, Volume 204, Issues 1-3, February 2001, Pages 221-232.

  "Reflux condenser mode with non-condensible gas: assessment of Cathare against Bethsy test 7.2C," B. Noel and R. Deruaz, Nuclear Engineering and Design, Volume 149, Issues 1-3, 1 September 1994, Pages 291-298.

- Secondary transient depressurization in one ROSA test (see the Y. Kukita, et al. paper, above) was found to cause a significant relocation of the primary mass. Condensate accumulated in the crossover leg as well as in the steam generator tubes of the affected loop. Reactor pressure vessel mass decreased during this time, and core dryout progressed until it was later slowed by an increase in the reflux flow rate.

The previously approved methodology for modeling the steam generator used a single thermal flow path to represent the PWR U-tubes. Therefore, the observed non-uniform behavior between different length U-tubes would not be predicted because of this simplification.

In response to RAI 5, AREVA has committed to performing studies on the modeling of the U-tubes in the Appendix K SBLOCA evaluation model to justify the licensing model to be used during the Design Certification review. Both multiple flow paths and axial details will be investigated for the U-tubes and the steam generator economizer regions.

The staff finds this approach acceptable for justifying the U.S. EPR steam generator model development for use in the Appendix K SBLOCA evaluation model.

3.4 Proposed Methodology for Non-LOCA Events

The U.S. EPR non-LOCA analyses will be performed with codes and methodologies with demonstrated capability and appropriate ranges of applicability to perform these transients. The S-RELAP5 code is the principal tool for U.S. EPR non-LOCA analyses.

The S-RELAP5 code includes many generic component models from which general systems can be simulated. These components include pumps, valves, pipes, heat releasing or absorbing structures, reactor point kinetics, electric heaters, turbines, separators, accumulators, and control and trip system components. Special process models are included for effects such as form loss, flow at an abrupt area change, branching, choked flow, counter-current flow limit (CCFL), boron tracking, and non-condensable gas transport. The reactor fuel is represented as a heat releasing or absorbing structure.

As discussed in Section 3.1, above, the staff finds there is reasonable assurance that the response of the U.S. EPR to non-LOCA transient and accidents can be evaluated with the currently approved S-RELAP5 code. However, the modeling of some features, notably the steam generators, is to be addressed by AREVA prior to, or during, the Design Certification review.

In response to RAI 8, AREVA has committed to performing studies on the modeling of the U-tubes and the economizer model to justify the licensing model to be used during the Design Certification review.

The staff finds this approach acceptable for justifying the U.S. EPR steam generator model development for use in non-LOCA (SRP Chapter 15) evaluations.

3.5 Neutronic Methodologies

Neutronics analysis falls within three basic design areas as noted in Section 3.2 above. The first design area is core reload and core follow analyses for which the predominant methodology is described in EMF-96-029(P)(A). This methodology uses the MICBURN/CASMO/PRISM code system when developing core models to evaluate steady-state core behavior with burnup at hot zero power or hot full power. The code system has been benchmarked to criteria based on those suggested in ANSI/ANS-19.6.1-1985, “Reload Startup Physics Tests for Pressurized Water Reactors.” In addition, the code system has been reviewed and accepted for use by the NRC for analysis of numerous core configurations (see Table 3-2 in ANP-10263).

The next design area is power distribution monitoring. The methodology for reconstructing the measured power distributions for comparison to calculated power distributions for plants that use the Aeroball Measurement System is discussed in ANP-10263, Appendix A, Section A.3. However, AREVA has not asked approval of inferred power distribution reconstruction methodology and uncertainty analysis in this TR. In response to RAI 22, AREVA has indicated that the power distribution measurement uncertainty associated with the Aeroball System will be submitted for NRC review and approval as part of a future topical report on setpoint safety analysis methods.

The final analysis area is safety analysis/thermal-hydraulic/thermo-mechanical support. As in the previous two cases, EMF-96-029(P)(A) provides the primary methodology for performing the calculations. In this case, branch calculations are typically performed from various points off the base core depletion. These generally consist of the calculation of core power distributions, reactivity parameters, rod/bank worths, critical boron concentrations, etc. at beginning-, middle-, and end-of-cycle (BOC, MOC, and EOC).

These parameters also allow for varying power levels, moderator temperatures, and xenon conditions. To support analyses of certain types of rapidly evolving transients (i.e., control rod ejection, main steam line break, uncontrolled rod/bank drop/withdrawal), without imposing the excessive conservatism inherent in a static reactor approximation, it is often necessary to employ three-dimensional transient techniques. For these cases, the TR BAW-10221P-A, “NEMO-K - A Kinetics Solution in NEMO” is applied. This methodology provides for modeling of neutron kinetics with transient modeling of the fuel rod and coolant temperatures to provide a more accurate prediction of core reactivity, power distributions, and rod/bank integral and differential worths.

The staff finds this approach acceptable for justifying the U.S. EPR use of current AREVA neutronic methods.

3.6 Applicability of Neutronic Codes/Methodologies to U.S. EPR

Topical report EMF-96-029(P)(A) provides the basis for AREVA’s PWR and U.S. EPR neutronic calculations and neutronics methodologies. TR BAW-10221P-A (NEMO-K) TR serves to augment these capabilities.

This methodology codes system was developed to be easily validated utilizing a standard set of validation criteria based on industry standards (ANSI/ANS-19.6.1-1985). The initial validation process included both Westinghouse and Combustion Engineering core designs with their differing fuel rod/assembly designs, core sizes, control rod designs, and in-core detectors (both moveable U235 fission chambers and fixed Rhodium detectors). The methodology has also been satisfactorily implemented on the Palisades reactor core configuration which incorporates BWR-style wide and narrow water gaps and cruciform control blades. Many of these cores include unique design characteristics which have been successfully modeled with the methodology. Examples of the applicability of the neutron methodology are provided in the ANP-10263.

4.0 CONCLUSIONS

The staff finds there is reasonable assurance that the currently approved neutronic codes and methods, CASMO-3G/MICRBURN-3, PRISM, LYNXT and NEMO-K, used by AREVA to evaluate typical 4-loop PWRs for core design/core follow, power distribution monitoring and safety analysis, are applicable to the U.S. EPR.

The staff finds there is reasonable assurance that the currently approved thermal-hydraulic code, S-RELAP5, used by AREVA to evaluate typical 4-loop PWRs for Appendix K-based SBLOCAs and for non-LOCA transients and accidents, is applicable to the U.S. PWR.

The staff finds the approach to be used to validate the steam generator model for use in the Appendix K-based SBLOCAs and for non-LOCA transients and accidents evaluations acceptable. These licencing analyses models will be justified prior to, or during, the Design Certification review.

cc:

Mr. Glenn H. Archinoff
AECL Technologies
481 North Frederick Avenue
Suite 405
Gaithersburg, MD 20877

Ms. Michele Boyd
Legislative Director
Energy Program
Public Citizens Critical Mass Energy
and Environmental Program
215 Pennsylvania Avenue, SE
Washington, DC 20003

Mr. Marvin Fertel
Senior Vice President
and Chief Nuclear Officer
Nuclear Energy Institute
1776 I Street, NW
Suite 400
Washington, DC 20006-3708

Mr. Ray Ganthner
AREVA, Framatome ANP, Inc.
3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506-0935

Dr. Charles L. King
Licensing Manager, IRIS Project
Westinghouse Electric Company
Science and Technology Department
20 International Drive
Windsor, CT 06095

Ms. Sherry McFaden
Framatome NP, Inc.
3315 Old Forest Road, OF-16
Lynchburg, VA 24501

Mr. John O'Neill
Pillsbury, Winthrop, Shaw, Pittman
2300 N Street, NW
Washington, DC 20037

Mr. Steve Seitz
AREVA
100 Dean Road
East Lyme, CT 06333

Mr. Robert E. Sweeney
IBEX ESI
4641 Montgomery Avenue
Suite 350
Bethesda, MD 20814

Mr. Gary Wright
Director
Division of Nuclear Facility Safety
Illinois Emergency Management Agency
1035 Outer Park Drive
Springfield, IL 62704

Email

APH@NEI.org (Adrian Heymer)
awc@nei.org (Anne W. Cottingham)
bob.brown@ge.com (Robert E. Brown)
BrinkmCB@westinghouse.com (Charles Brinkman)
carey.fleming@constellation.com (Carey Fleming)
chris.maslak@ge.com (Chris Maslak)
cwaltman@roe.com (C. Waltman)
david.hinds@ge.com (David Hinds)
david.lewis@pillsburylaw.com (David Lewis)
dlochbaum@UCSUSA.org (David Lochbaum)
erg-xl@cox.net (Eddie R. Grant)
frankq@hursttech.com (Frank Quinn)
gcesare@enercon.com (Guy Cesare)
greshaja@westinghouse.com (James Gresham)
james.beard@gene.ge.com (James Beard)
jcurtiss@winston.com (Jim Curtiss)
jgutierrez@morganlewis.com (Jay M. Gutierrez)
jim.riccio@wdc.greenpeace.org (James Riccio)
JJNesrsta@cpsenergy.com (James J. Nesrsta)
john.o'neil@pillsburylaw.com (John O'Neil)
Joseph.savage@ge.com (Joseph Savage)
Joseph_Hegner@dom.com (Joseph Hegner)
junichi_uchiyama@mhi.co.jp (Junichi Uchiyama)
KSutton@morganlewis.com (Kathryn M. Sutton)
kwaugh@impact-net.org (Kenneth O. Waugh)
lynchs@gao.gov (Sarah Lynch - Meeting Notices Only)
maria.webb@pillsburylaw.com (Maria Webb)
mark.beaumont@wsms.com (Mark Beaumont)
matias.travieso-diaz@pillsburylaw.com (Matias Travieso-Diaz)
media@nei.org (Scott Peterson)
mike_moran@fpl.com (Mike Moran)
mwetterhahn@winston.com (M. Wetterhahn)
mwl@nei.org (Melanie Lyons)
patriciaL.campbell@ge.com (Patricia L. Campbell)
paul.gaukler@pillsburylaw.com (Paul Gaukler)
Petrovb@westinghouse.com (Bojan Petrovic)
PGunter@NIRS.org (Paul Gunter)
pshastings@duke-energy.com (Peter Hastings)
RJB@NEI.org (Russell Bell)
RKTemple@cpsenergy.com (R.K. Temple)
roberta.swain@ge.com (Roberta Swain)
rod.krich@constellation.com (Mr. Rod Krich)
Ronda.Daflucas@areva.com (Ronda Daflucas)
ronald.hagen@eia.doe.gov (Ronald Hagen)
sandra.sloan@areva.com (Sandra Sloan)
sfrantz@morganlewis.com (Stephen P. Frantz)
steven.hucik@ge.com (Steven Hucik)
tkkibler@scana.com (Tria Kibler)

tom.miller@hq.doe.gov (Tom Miller)
tom.miller@nuclear.energy.gov (Thomas P. Miller)
trsmith@winston.com (Tyson Smith)
vijukrp@westinghouse.com (Ronald P. Vijuk)
waraksre@westinghouse.com (Rosemarie E. Waraks)
wayne.marquino@ge.com (Wayne Marquino)
whorin@winston.com (W. Horin)