September 24, 2008

Mr. William R. Campbell, Jr.
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNIT 1 – ISSUANCE OF AMENDMENT REGARDING CORE OPERATING LIMITS REPORT REFERENCES FOR REALISTIC LARGE BREAK LOSS-OF-COOLANT ACCIDENT METHODOLOGY (TAC NO. MD8532)

Dear Mr. Campbell:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 320 to Facility Operating License No. DPR-77 for the Sequoyah Nuclear Plant, Unit 1 (SQN-1). The amendment consists of changes to the technical specifications (TSs) in response to your application dated April 14, 2008 (TS-08-01).

The amendment revises TS 6.9.1.14.a, by adding the best estimate large break loss-of-coolant accident (LOCA) analysis described in the AREVA Topical Report EMF-2103P-A, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," as an acceptable methodology for use in the SQN-1 Core Operating Limits Report.

A copy of the safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Brendan T. Moroney, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-327

Enclosures:
1. Amendment No. 320 to License No. DPR-77
2. Safety Evaluation

cc w/enclosures: See next page

September 24, 2008

Mr. William R. Campbell, Jr.
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNIT 1 – ISSUANCE OF AMENDMENT REGARDING CORE OPERATING LIMITS REPORT REFERENCES FOR REALISTIC LARGE BREAK LOSS-OF-COOLANT ACCIDENT METHODOLOGY (TAC NO. MD8532)

Dear Mr. Campbell:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 320 to Facility Operating License No. DPR-77 for the Sequoyah Nuclear Plant, Unit 1 (SQN-1). The amendment consists of changes to the technical specifications (TSs) in response to your application dated April 14, 2008 (TS-08-01).

The amendment revises TS 6.9.1.14.a, by adding the best estimate large break loss-of-coolant accident (LOCA) analysis described in the AREVA Topical Report EMF-2103P-A, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," as an acceptable methodology for use in the SQN-1 Core Operating Limits Report.

A copy of the safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Brendan T. Moroney, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-327

Enclosures:
1. Amendment No. 320 to License No. DPR-77
2. Safety Evaluation

cc w/enclosures: See next page

DISTRIBUTION:

PUBLIC	LPL2-2 R/F	RidsOgcRp	RidsNrrDorlDpr
RidsAcrsAcnw&mMailCenter	RidsNrrDorlLpl2-2	RidsNrrLACSola	RidsNrrPMBMoroney
RidsRgn2MailCenter	RidsNrrDssSrxb	RidsNrrDirsItsb	RidsNrrDssScvb
RLobel, NRR	FOrr, NRR	RidsNrrPMTOrf	

ADAMS Accession Number: LTR:ML082470628 PKG:ML082470382 TS:ML082470390 NRR-058

OFFICE	LPL2-2/PM	LPL2-2/LA	ITSB/BC	SCVB/BC	SRXB/BC	OGC	LPL2-2/BC
NAME	BMoroney	RSola	RElliott	RDennig By memo dated	GCranston By memo dated	NLO w/ comments RHolmes	TBoyce
DATE	09/11/08	09/11/08	09/12/08	08/19/08	7 / 18 /08	09/19/08	09/24/08

OFFICIAL RECORD COPY

William R. Campbell, Jr.
Tennessee Valley Authority
cc:
Mr. Ashok S. Bhatnagar
Senior Vice President
Nuclear Generation Development
and Construction
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Vice President
Nuclear Support
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. H. Rick Rogers
Vice President
Nuclear Engineering & Technical Services
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Timothy P. Cleary, Site Vice President
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37384-2000

General Counsel
Tennessee Valley Authority
6A West Tower
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. John C. Fornicola, Manager
Nuclear Assurance
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Ms. Beth A. Wetzel, Manager
Corporate Nuclear Licensing and
Industry Affairs
Tennessee Valley Authority
4X Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

SEQUOYAH NUCLEAR PLANT

Mr. James D. Smith, Manager
Licensing and Industry Affairs
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37384-2000

Mr. Christopher R. Church, Plant Manager
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37384-2000

Senior Resident Inspector
Sequoyah Nuclear Plant
U.S. Nuclear Regulatory Commission
2600 Igou Ferry Road
Soddy Daisy, TN 37379

Mr. Lawrence E. Nanney, Director
TN Dept. of Environment & Conservation
Division of Radiological Health
Third Floor, L and C Annex
401 Church Street
Nashville, TN 37243-1532

County Mayor
Hamilton County Courthouse
Chattanooga, TN 37402-2801

Mr. Larry E. Nicholson, General Manager
Performance Improvement
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Michael A. Purcell
Senior Licensing Manager
Nuclear Power Group
Tennessee Valley Authority
4X Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Ms. Ann P. Harris
341 Swing Loop Road
Rockwood, TN 37854

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 320
License No. DPR-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated April 14, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the technical specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-77 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 320, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas H. Boyce, Chief
Plant Licensing Branch II-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to License No. DPR-77 and
the Technical Specifications

Date of Issuance: September 24, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 320

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace page 3 of License No. DPR-77 with the attached revised page 3.

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
6-13a	6-13a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 320 TO FACILITY OPERATING LICENSE NO. DRP-77

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-327

1.0 INTRODUCTION

By letter dated April 14, 2008 (Ref. 1), the Tennessee Valley Authority (TVA), the licensee for Sequoyah Nuclear Plant, Unit 1 (SQN-1) requested an amendment to the SQN-1 Operating License DPR-77 to add a new reference in Section 6.9.1.14.a of the Technical Specifications (TSs). The new reference is AREVA Topical Report EMF-2103P-A, "Realistic Large Break LOCA [Loss-Of-Coolant Accident] Methodology for Pressurized Water Reactors" (EMF-2103P-A) (Ref. 2). Enclosure 2 to Ref. 1 provided the SQN-1 evaluation of the large break LOCA (LBLOCA), as documented in AREVA Topical Report ANP-2695P, Revision 0, dated February 2008 (ANP-2695P) (Ref.3).

The U.S. Nuclear Regulatory Commission (NRC or Commission) staff reviewed the licensee's demonstration evaluations of the emergency core cooling system (ECCS) performance analyses, done in accordance with the AREVA best estimate (BE) LBLOCA methodology, with SQN-1 operating at its currently licensed core power of 3455 megawatt thermal (Mwt). These specific analyses were performed to demonstrate that the SQN-1 plant meets NRC requirements and the criteria of Section 50.46 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.46). Also, these specific analyses, when approved herein, will be acceptable and, specifically, applicable to SQN-1 operated with the fuels identified in Table 1 of this safety evaluation (SE). The BE LBLOCA analyses for SQN-1 were conducted assuming that the plant uses cores containing M5 clad uranium oxide fuel assemblies.

One aspect of the proposed revision is the calculation of the containment pressure during the postulated LOCA. The NRC staff also reviewed the calculation of containment pressure during the postulated LOCA.

2.0 REGULATORY EVALUATION

2.1 Containment Analysis

Standard Review Plan, Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," provides guidance on assumptions made regarding the operation of engineered safety feature containment heat removal systems and the effectiveness of structural heat sinks within containment to remove energy from the containment atmosphere. The objective of this guidance is to obtain a conservatively large heat removal rate from the containment atmosphere in order to minimize the containment pressure. The accompanying Branch Technical Position (BTP) 6-2, "Minimum Containment Pressure Model for PWR [Pressurized Water Reactor] ECCS Performance Evaluation," provides detailed guidance to achieve the same objective. The licensee has treated some parameters using the guidance of BTP 6-2 and other parameters statistically using the guidance of Regulatory Guide (RG) 1.157.

RG 1.157 (issued May 1989), "Best Estimate Calculations of Emergency Core Cooling System Performance," which the licensee uses as guidance for the realistic (R) LBLOCA analyses, states that:

> The containment pressure used for evaluating the post-blowdown phase of the loss-of-coolant accident should be calculated in a best estimate manner and should include the effects of containment heat sinks. The calculation should include the effects of operation of all pressure reducing equipment assumed to be available. Best estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis. (Section C, Paragraph 3.12.1)

RG 1.157 also states (Section C.1) that the introduction of conservative bias is acceptable as long as this does not result in unrealistic results or omit important phenomena.

2.2 LBLOCA Analysis

The BE LBLOCA analyses were performed to demonstrate that the ECCS design would provide sufficient ECCS flow to transfer the heat from the reactor core following an LBLOCA at a rate such that: (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) the clad metal-water reaction would be limited to less than the amounts that would compromise cladding ductility and result in excessive hydrogen generation.

The NRC staff reviewed the analyses to assure that the safety functions could be accomplished, assuming a single failure, for LBLOCAs and considering the availability of only onsite or offsite electric power (i.e., assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available, with offsite electric power available).

The NRC staff used the acceptance criteria for ECCS performance provided in 10 CFR 50.46, in assessing the application of the AREVA RLBLOCA methodology for SQN-1. In its assessment of the application of the methodology for SQN-1, the NRC staff also reviewed the limitations and conditions stated in its SE (Ref. 4) supporting generic approval of the AREVA RLBLOCA methodology and the range of parameters described in the EMF-2103P-A.

3.0 TECHNICAL EVALUATION

3.1 Evaluation of Containment Pressure Calculation

The SQN-1 RLBLOCA analysis assumes a break in the cold leg piping between the reactor coolant pump and the reactor vessel for the reactor coolant system loop containing the pressurizer. ANP-2695P states that this assumption is based on deterministic studies.

Considering the assumed single failure, one charging pump, one safety injection pump and one residual heat removal pump are assumed to be operating. In addition, ANP-2695P assumes both containment spray pumps are operating. This complies with the guidance of RG 1.157 that all containment heat removal equipment should be considered operating.

The SQN-1 Updated Final Safety Analysis Report (UFSAR), Section 6.2.2.1, states that the SQN-1 containment heat removal system consists of the following:

1. Ice Condenser
2. Air Return Fan System
3. Containment Spray System
4. Residual Heat Removal Spray System

Of these, only the ice condenser and the containment spray system are important in the time frame of the calculation of the peak cladding temperature (PCT) and core quenching. (The SQN-1 design does not include fan coolers in its containment pressure control.)

The assumed initial ice mass is 2.448 million pounds. This is approximately 10 percent more than the minimum amount allowed by the SQN-1 TSs. Figure 6-26 of ANP-2695P shows that approximately 80 percent of the ice mass remains at 600 seconds. Therefore, the ice mass used in the ICECON simulation is adequate for the prediction of containment pressures during the LOCA.

The start time of the air recirculation fan system is 600 seconds. The licensee states in Section 3.3 of ANP-2695P that:

> Because the start time of the recirculation fan is 600 seconds, forced flow from the upper compartment to the lower compartment is not likely to occur during the time period analyzed.

As stated above, both trains of the containment spray system are assumed to operate. The containment spray pumps take suction from the refueling water storage tank (RWST). The RWST water temperature is conservatively assumed to be 55 degrees Fahrenheit (°F). (The RWST water temperature is conservatively taken as 110 °F for ECCS injection into the vessel.) In addition, the licensee assumes an early start to the containment spray flow (eight seconds after the containment safety injection signal). Both pumps are activated at 100-percent capacity (7700 gallons per minute). Since the containment spray pump model is conservative (both trains operating, low water temperature, early start of flow), the NRC staff finds the containment spray modeling to be acceptable.

The residual heat removal spray system is only needed after ice melt has occurred and is, therefore, not important for these analyses.

The NRC has previously approved the realistic LOCA containment modeling for a PWR with a subatmospheric containment (Ref. 5). The NRC has also previously approved the realistic LOCA modeling for SQN-2. In response to a staff request for additional information (RAI) during the SQN-2 review (Ref. 6), the licensee stated that the general aspects of the approach approved by the NRC for the subatmospheric containment are the same as those for SQN-2. Section 6.9 of ANP-2695P provides responses to the same RAIs for SQN-1. These responses demonstrate that the responses for SQN-2 are applicable to SQN-1. The RLBLOCA approach uses both realistic and conservative modeling assumptions. The dominant phenomena influencing the containment response are the initial pressure and volume, heat transfer to internal structures, break size and effluent modeling. The break size and effluent modeling are not addressed in this SE. As stated in Ref. 3, PIRT (Phenomena Identification and Ranking Table) analyses performed by AREVA indicate that two containment-related phenomena directly influence peak cladding temperature: the initial containment pressure and initial containment temperature

The containment pressure is calculated using values of the containment volume that are sampled over a range from a nominal volume to the containment empty volume. The containment empty volume is the volume with no internal equipment or structures included. Since the pressure decreases with increasing containment volume, this range ensures that the selection of the containment volume is either best estimate (if the nominal containment volume is sampled) or conservative (that is, results in a pressure lower than nominal). The initial containment atmosphere temperature ranges between 80 °F to 100 °F for the upper compartment and 95 °F to 130 °F for the lower compartment. The temperature is important in determining the air mass within the containment, which affects the containment pressure. These temperature ranges are typical and acceptable. Both variables are assumed to have uniform distributions over their respective ranges.

Table 3-9 of ANP 2695P, “Passive Heat Sinks in Containment,” lists the properties of the structures in containment credited as heat sinks. The licensee used properties of passive heat sinks biased flow for another application. In order for these values to be conservative for the realistic LOCA application, the licensee increased the surface area of these heat structures by 10 percent to provide further margin toward low pressures. Since these data are conservative for containment minimum pressure calculations, they are acceptable for the realistic LOCA calculations since the guidance of RG 1.157 permits conservative input as long as the use of such information does not make the analysis unrealistic. It is the NRC staff’s judgment that the use of this passive heat sink information does not make the analysis unrealistic.

The heat transfer coefficients for determining the heat transfer between the containment atmosphere and the passive heat sinks are important. The realistic LOCA methods use the Uchida condensing heat transfer correlation with a multiplier to account for the greater amount of heat transfer that would occur during the blow-down phase of the LOCA. The NRC staff has previously found this acceptable when justified on a plant-specific basis since the value of the multiplier could change with each analysis. ANP-2695P states that the multiplier used was specifically validated for use in the SQN-1 RLBLOCA analyses. The licensee stated that further conservatism was provided since surface coatings, where they existed, were modeled as the underlying material so that the insulating effect on a metallic structure was discounted, and, as a further pressure lowering measure, condensing heat transfer was modeled on those structures which fall below the transient water level during the LOCA.

An important part of the containment pressure calculation for an ice condenser containment such as SQN-1's is the heat transfer between the containment steam and the ice melt. The licensee states that the water spillage plus drainage from the ice chest falls through the lower containment vapor. This condenses steam and reduces the containment pressure. The ice chest drainage flow is treated as 100-percent efficient spray during the post-blowdown period of the transient. This modeling is conservative and acceptable.

The containment pressure is calculated by S-RELAP using containment models derived from the ICECON computer code. The NRC staff has previously approved these methods for minimum pressure calculations (Ref. 7). It is noted that ICECON is based on the CONTEMPT computer code. An NRC staff study dated August 31, 2002 (Ref. 8), showed that the CONTEMPT code containment spray model "tends to reduce pressure more rapidly than the data indicates [sic]." Therefore, the containment spray model is acceptable for containment pressure calculations.

The staff finds the licensee's referencing and use of the methods of EMF-2103P-A, Revision 0, acceptable with respect to the determination of containment pressure since the calculation models have been previously approved, the calculations use best estimate or conservative input values and models, the licensee has demonstrated the acceptability of the multiplier used with the Uchida heat transfer correlation and, in general, the containment pressure calculation is in conformance with the guidance of RG 1.157 in that the calculations are best estimate or not so conservative as to be unrealistic.

3.2 Evaluation of AREVA BE LBLOCA Analyses Methodology

The NRC staff's review of the acceptability of the AREVA BE LBLOCA methodology for SQN-1 focused on assuring that the licensee and its vendor have processes to assure that specific input parameters or bounding values and ranges (where appropriate) are used to conduct the SQN-1 LBLOCA analyses, that the analyses will be conducted within the conditions and limitations of the NRC approved AREVA BE LBLOCA methodology, and that the results will satisfy the requirements of 10 CFR 50.46(b) for SQN-1 operating at its present licensed power.

Ref. 3 contains the statement, "Both entities [TVA and AREVA] have ongoing processes that assure the ranges and values of input parameters for the Sequoyah Unit 1 Station RLBLOCA analysis bound those of the as-operated plant." The staff finds that this statement, along with the generic acceptance of the AREVA RLBLOCA analysis methodology, provides assurance that the AREVA RLBLOCA analysis methodology and its LBLOCA analyses apply to SQN-1 operated at its current licensed power level.

The licensee provided the results for the SQN-1 RLBLOCA analyses, operating at the rated power of 3455 Mwt (performed in accordance with the AREVA RLBLOCA methodology). The licensee's results for the calculated peak cladding temperatures (PCTs), the maximum cladding oxidations (local), and the maximum core-wide cladding oxidations for SQN-1 are provided in the following table along with the acceptance criteria of 10 CFR 50.46(b). The licensee's analyses also assume credit for the increased accuracy of a leading edge flow meter device in the determination of the reactor operating power assumed in the LBLOCA analyses.

SEQUOYAH UNIT 1
LARGE BREAK LOCA ANALYSIS RESULTS

Parameter	SQN-1 Realistic LBLOCA Results	10 CFR 50.46 Limits
Limiting Break Size/Location	3.078 ft^2/side Split/PD*	N/A
Cladding Material	M-5	(Cylindrical) Zircaloy , Zirlo, (or M-5**)
Peak Clad Temperature	1809 ^oF	2200 ^oF (10 CFR 50.46(b)(1))
Maximum Local Oxidation	1.86%	17.0% (10 CFR 50.46(b)(2))
Maximum Total Core-Wide Oxidation (All Fuel)	< 0.0432 %	1.0% (10 CFR 50.46(b)(3))

* Split/PD Split break at the pump discharge.
** M-5 is equivalent to Zircaloy and ZIRLO in applications of 10 CFR 50.46 criteria.

The power assumed in the analyses (3479 Mwt) is approximately 0.7 percent higher than the operating power (3455 Mwt) of SQN-1 to account for measurement uncertainties. The licensee's LBLOCA analyses for SQN-1 properly assumed operation at a constant core power, and did not range reactor core power. The NRC finds this acceptable.

The analyses addressed the availability of offsite power correctly by ranging each case separately. This is acceptable because it satisfies General Design Criterion 35 of 10 CFR 50 Appendix A, in that each distribution type has been accounted for separately with its own set of cases, thereby addressing possible concerns associated with the mixing of two separate statistical spectra. Therefore, the NRC finds this treatment acceptable as discussed.

On August 3, 1998, the NRC issued Information Notice 98-29: "Predicted Increase in Fuel Rod Cladding Oxidation," expressing concern that predicted cladding total oxidation (including both pre-accident and accident oxidation) resulting from a postulated LOCA could for some plants exceed the 17 percent limit, and that LOCA methodologies were not addressing that concern.

In letters dated March 31 and November 8, 1999, to the Nuclear Energy Institute, NRC provided its position that both pre-accident and accident oxidation must be estimated, citing several references, including the Opinion of the Commission dated December 28, 1973, that demonstrate that the NRC position regarding oxidation predates the present LOCA acceptance criteria and the first accepted LOCA evaluation models under those criteria.

Therefore, the NRC staff considers the licensee's estimate of oxidation resulting from the postulated LBLOCA alone low for M-5 cladding (1.86 percent, without considering pre-LOCA oxidation and without considering oxidation on both inside and outside surfaces). However, the low calculated oxidation level is, at least in part, attributable to the lower calculated LBLOCA peak cladding temperature (PCT, 1809 °F) associated with the replacement of the SQN-1 steam generators. The low calculated total core-wide oxidation for LBLOCA is also likely the result of the LBLOCA analysis methodology's consideration of only fresh fuel assemblies in the analyses and the treatment of radiation heat transfer.

However, the NRC staff also considers that, even if the calculated PCT and local oxidation were reasonably greater (less than 100 °F and 10 percent greater, respectively) than reported above, the total expected oxidation, including all factors which should have been considered, would likely be less than the 17 percent limit specified in the regulation.

The concern with core-wide oxidation relates to the amount of hydrogen generated during a LOCA. Because hydrogen that may have been generated pre-LOCA (during normal operation) will be removed from the reactor coolant system throughout the operating cycle, the NRC staff noted that pre-existing oxidation does not contribute to the amount of hydrogen generated post-LOCA and therefore it does not need to be addressed further when determining whether the calculated total core-wide oxidation meets the 1.0 percent criterion of 10 CFR 50.46(b)(3).

As discussed previously, TVA had AREVA perform a BE LBLOCA analysis for SQN-1 operating at the licensed power level of 3455 MWt using an NRC-approved AREVA BE LBLOCA analysis methodology. The staff concludes that the above discussions provide assurance of the applicability of the AREVA BE LBLOCA methodology to SQN-1 and support the validity of the conclusion that at the analyzed licensed power level, the SQN-1 core will be amenable to cooling as required by 10 CFR 50.46(b)(4). The capability of SQN-1 to satisfy the long term cooling requirements of 10 CFR 50.46(b)(5) is unaffected by this amendment.

3.3 SQN-1 TS Change

The licensee proposed to revise TS 6.9.1.14.a to reflect use of a new LBLOCA analysis methodology to perform LBLOCA analyses in support of SQN-1 operation. Specifically, TS Page 6-14, "CORE OPERATING LIMITS REPORT (continued)," would be amended to add EMF-2103P-A as the licensing basis LBLOCA methodology for the SQN-1 plant. The NRC staff reviewed the TS provision, assessed it for consistency against NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 3 (Ref. 9), as stated below, and found its content acceptable and compatible with a proposed COLR (when the COLR change is submitted, including the reference as discussed below).

For the proposed change, the following reference will be added TS 6.9.1.14.a:

9. EMF-2103P-A, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors."

This methodology was found to apply to all conventional Westinghouse and Combustion Engineering PWR designs in the NRC generic SE of the EMF-2103P-A methodology. Therefore, the realistic LBLOCA methodology described in EMF-2103P-A is acceptable for application to SQN-1, which is a PWR of Westinghouse design, and for inclusion in TSs for the SQN-1 plant. The proposed change does not include the EMF-2103P-A revision number or the date of approval for the methodology. However, the licensee will list the topical report, including the latest revision number used at SQN-1 and date of approval, in the COLR for SQN-1, consistent with guidance provided in NUREG-1431.

The NRC staff finds that the EMF-2103P-A methodology is applicable to SQN-1, and that the limitations and conditions of the NRC's SE approving the EMF-2103P-A methodology were satisfied, for the present SQN-1 operating power. Therefore, the NRC staff concludes that the proposed addition of EMF-2103P-A to SQN-1 TS 6.9.1.14.a is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official, Ms. Elizabeth Flanagan of the Tennessee Bureau of Radiological Health, was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (72 FR 49583). Accordingly, the amendment meets 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 amendment.

6.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter, TVA to NRC, Sequoyah Nuclear Plant (SQN) Unit 1, Technical Specifications (TS) Change 08-01, "Revision of Core Operating Limits Report (COLR) References for Realistic Large Break Loss-of Coolant Accident Methodology," April 14, 2008, (ADAMS

[Agencywide Documents Access and Management System] Accession No. ML081080337).

2. AREVA Topical Report EMF-2103P-A, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” April 2003.

3. AREVA (Framatome) Topical Report ANP-2695P, Revision 0, Sequoyah Nuclear Plant Unit 1 Realistic Large Break Loss-of-Coolant Accident Analysis ANP-2695P, Revision 0,” February 2008.

4. NRC Safety Evaluation, “Safety Evaluation on Framatome ANP Topical Report EMF-2103P, Revision 0, Realistic Large Break Loss-of-Coolant Methodology for Pressurized Water Reactors,” April 9, 2003.

5. Letter from US NRC to David A Christian, Sr. Vice President and Chief Nuclear Officer, Virginia Electric Power Company, North Anna Power Station Unit 1 – Issuance of Amendment Re: Use of Framatome ANP Advanced Mark-BW Fuel, August 20, 2004 (ADAMS Accession No. ML042330659).

6. Letter from James D. Smith, Manager, Site Licensing and Industry Affairs, Tennessee valley Authority, Sequoyah Unit 2- response to request for additional information (RAI) Regarding Large Break Loss-of-Coolant Accident Analysis Methods, Response to RAI 9(e), December 21, 2007 (ADAMS Accession No. ML073601002).

7. Letter from US NRC to David A Christian, Sr. Vice president and Chief Nuclear Officer, Virginia Electric Power Company, North Anna Power Station Unit 1 – Issuance of Amendment Re: Use of Framatome ANP Advanced Mark-BW Fuel, August 20, 2004 (ADAMS Accession No. ML042330659).

8. Jack Tills, Allen Notafrancesco, Ken Murata, “CONTAIN Code Qualification Report/User Guide for Auditing Design Basis PWR Calculations,” SMSAB-02-3, U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, August 31, 2002, (ADAMS Accession Number ML022490371).

9. NUREG-1431, “Standard Technical Specifications, Westinghouse Plants,” Revision 3, TS 5.6.5.

Principal Contributors: Richard Lobel
Frank Orr

Date: September 24, 2008