Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

March 17, 2000

Template NRR-058

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING BEST ESTIMATE LARGE BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS, TS-98-016 (TAC NO. MA6038)

Dear Mr. Scalice:

The Commission has issued the enclosed Amendment No. 21 to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant, Unit 1. This amendment is in response to your application dated June 25, 1999, as supplemented January 25, 2000. Your letters requested approval to apply the Westinghouse generic best estimate large break loss-of-coolant accident analysis methodology, using the WCOBRA/TRAC code to the Watts Bar Unit 1 plant. As discussed in the enclosed safety evaluation, this completes the staff's activities associated with TAC Numbers MA6038, M89477 and M93767.

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/

Robert E. Martin, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosures: 1. Amendment No. 21 to NPF-90
2. Safety Evaluation

cc w/enclosures: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 17, 2000

Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
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Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager, Section 2
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cc w/enclosures: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-390

WATTS BAR NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 21
License No. NPF-90

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 25, 1999, as supplemented January 25, 2000, 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 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. NPF-90 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 21 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, to be implemented prior to startup following the Unit 1, Cycle 3 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard P. Correia

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Project Licensing Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 17, 2000

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ATTACHMENT TO AMENDMENT NO. 21

FACILITY OPERATING LICENSE NO. NPF-90

DOCKET NO. 50-390

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

Remove Pages	Insert Pages
3.5-2	3.5-2
5.0-32	5.0-32
B 3.2-2	B 3.2-2
B 3.2-4	B 3.2-4
B 3.2-13	B 3.2-13
B 3.2-14	B 3.2-14
B 3.5-2	B 3.5-2
B 3.5-3	B 3.5-3
B 3.5-4	B 3.5-4
B 3.5-12	B 3.5-12
B 3.5-13	B 3.5-13

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 7630 gallons and ≤ 8000 gallons.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 610 psig and ≤ 660 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 2400 ppm and ≤ 2700 ppm.	31 days AND -----NOTE------ Only required to be performed for affected accumulators --------------- Once within 6 hours after each solution volume increase of ≥ 75 gallons, that is not the result of addition from the refueling water storage tank

(continued)

5.9 Reporting Requirements (continued)

5.9.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to the initial and each reload cycle, or prior to any remaining portion of a cycle, and shall be documented in the COLR for the following:

LCO 3.1.4 Moderator Temperature Coefficient
LCO 3.1.6 Shutdown Bank Insertion Limit
LCO 3.1.7 Control Bank Insertion Limits
LCO 3.2.1 Heat Flux Hot Channel Factor
LCO 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor
LCO 3.2.3 Axial Flux Difference
LCO 3.9.1 Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalphy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration.

2a. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," March 1998 (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

b. WCAP-10054-P-A, "Small Break ECCS Evaluation Model Using NOTRUMP Code," August 1985, Addendum 2, Rev. 1: "Addendum to the Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997. (W Proprietary). (Methodology for Specifications 3.2.1 - Heat FTux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

3. WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).(Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT," April 1995. (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

(continued)

BASES

BACKGROUND (continued)	the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES

LCO (continued)

The expression for $F^W_Q(Z)$ is:

$$F^W_Q(Z) = F^C_Q(Z)\ W(Z)$$

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during a small break LOCA, and assures with a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1).

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F^C_Q(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F^C_Q(Z)$ is $F^M_Q(Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F^M_Q(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the

(continued)

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 3);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to a value which satisfies the 95/95 criterion for the DNB correlation used. Refer to the Bases for the Reactor Core Safety Limits, B 2.1.1, for a discussion of the applicable DNBR limits. The W-3 Correlation with a DNBR limit of 1.3 is applied in the heated region below the first mixing vane grid. In addition, the W-3 DNB correlation is applied in the analysis of accident conditions where the system pressure is below the range of the WRB-1 correlation. For system pressures in the range of 500 to 1000 psia, the W-3 correlation DNBR limit is 1.45 instead of 1.3.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Application of these criteria provides assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3), model $F^N_{\Delta H}$ as well as the Nuclear Heat Flux Hot Channel Factor (FQ(Z)).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.7, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor (FQ(Z))."

$F^N_{\Delta H}$ and FQ(Z) are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of the NRC Policy Statement.

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat

(continued)

BASES

BACKGROUND (continued)

This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. Although not required for accident mitigation, the valves will automatically open as a result of an SI signal. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is also considered to determine if it yields limiting results. The loss of offsite power assumption imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The limiting large break LOCA is a double ended guillotine break in the cold leg. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and centrifugal charging pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46, Paragraph b (Ref. 3) will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. The safety analyses support a range of 7518 gallons to 8191 gallons. To allow for instrument inaccuracy, values of 7630 gallons and 8000 gallons are specified.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The small break LOCA analysis is performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure analysis limit of 690 psig prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity. The LOCA analyses support a range of 585 to 690 psig. To account for the accumulator tank design pressure rating, and to allow for instrument accuracy values of ≥ 610 psig and ≤ 660 psig are specified for the pressure indicator in the main control room.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 4).

(continued)

BASES

BACKGROUND (continued)

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence for a loss of offsite power. If offsite power is available, the safeguard loads start immediately. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46, Paragraph b (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with or without loss of offsite power and with a single failure disabling one ECCS train (in the containment pressure analysis, both EDG trains are conservatively assumed to operate due to requirements for modeling full active containment heat removal system operation); and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

UNITED STATES

NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 21 TO FACILITY OPERATING LICENSE NO. NPF-90

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-390

1.0 INTRODUCTION

By letter dated June 25, 1999, as supplemented January 25, 2000, the Tennessee Valley Authority (TVA, the licensee) submitted a request for changes to the Watts Bar Nuclear Plant, Unit 1 (WBN1), Technical Specifications (TS). The requested changes are associated with the use of the Westinghouse (W) best estimate (BE) large break (LB) loss-of-coolant accident (LOCA) analyses methodology. The January 25, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original notice.

2.0 EVALUATION

2.1 Best Estimate Large Break LOCA (LBLOCA) Methodology

The TVA June 25, 1999, submittal discusses LBLOCA re-analyses to reflect the use of the W BE LOCA model to perform the re-analyses, the LBLOCA results, and TS changes reflecting use of the BE LOCA methodology. TVA used the approved W BE LBLOCA methodology discussed in WCAP-12945 P-A March 1998 to perform the WBN1 LBLOCA analyses discussed in its submittal. The sample calculations presented in WCAP-12945 P-A are for 3- and 4-loop W plant designs. WBN1 is a 4-loop W plant. Accordingly, the Nuclear Regulatory Commission (NRC) staff concludes that TVA has demonstrated that the WCAP-12945-P-A methodology is applicable to WBN1.

2.2 Small Break (SB) LOCA Analysis Methodology

The TVA June 25, 1999, submittal also discusses SBLOCA re-analyses to reflect the use of the W NOTRUMP/COSI SBLOCA model to perform the re-analyses, the SBLOCA results, and TS changes reflecting use of the NOTRUMP/COSI SBLOCA methodology. TVA used the approved W SBLOCA methodology discussed in WCAP-10054 P-A, Addendum 2, Revision 1, July 1997, to perform the WBN1 SBLOCA analyses discussed in its submittal. The sample calculations presented in WCAP-10054 P-A are for W plant designs. WBN1 is a 4-loop W plant. Accordingly, the NRC staff concludes that TVA has demonstrated that the WCAP-10054-P-A methodology is applicable to WBN1.

ENCLOSURE

2.3 LOCA Analysis Input Values

In a letter dated January 25, 2000, the licensee stated that TVA/W ongoing processes assure that the LOCA analysis input values for peak cladding temperature (PCT)-sensitive parameters conservatively bound the as-operated plant values for those parameters. From this the NRC staff concludes that the W BE LBLOCA analysis methodology described in WCAP-12945 P-A and the W NOTRUMP/COSI SBLOCA analysis methodology apply to the WBN1 plant. The licensee has shown that these methodologies apply to WBN1 by considering the plant's design and by indicating that TVA/W processes assure appropriate analysis input values. The licensee has performed new analyses of record implementing these methodologies and processes.

2.4 WBN1 LOCA Analyses

The sections of the licensee's June 25, 1999, submittal discussed above also describe LBLOCA licensing analyses performed by the licensee to establish new analyses of record for licensing uses, such as showing compliance with the requirements of Title 10, *Code of Federal Regulations* (10 CFR), Part 50, Section 46, establishing WBN1 TS and surveillance requirements, and for reference in complying with the reporting requirements of the governing regulations. The calculated PCT is 1892 ° F; the calculated maximum cladding oxidation (including pre-accident and transient oxidation) is 15 percent; and the maximum hydrogen generation is 0.61 percent. The analyses indicate that the core remains amenable to cooling and that long-term cooling is maintained. These results conform with the criteria provided in 10 CFR 50.46(b).

2.5 TS and Other Licensing Uses

In the preceding sections the NRC staff has concluded that the LOCA methodologies and analyses discussed in the licensee's submittal dated June 25, 1999 are acceptable. Therefore, the NRC staff concludes that they are acceptable for inclusion in licensing documentation, including the WBN1 Updated Final Safety Analysis Report ((U)FSAR), TS, and WBN1 core operation limits report (COLR).

The licensee's submittal also proposed several TS changes to reflect use of the BE LBLOCA methodology and use of the W small break (SB) LOCA analysis methodology described in WCAP-10054 P-A, July 1997, which is generically approved for application to W plant designs.

The licensee's proposed changes to TS are:

a. TS Surveillance Requirement 3.5.1.2 is changed to reflect the required minimum and maximum accumulator borated water volumes. This change is acceptable because it is based on LBLOCA analyses performed with an approved LBLOCA analysis methodology. And,

b. TS 5.9.5, Core Operating Limits Report, is changed to reflect that the approved licensing basis LB and SB LOCA analysis methodologies are as described in WCAP-12945 P-A, March 1998 and WCAP-10054 P-A, July 1997, respectively. These references are acceptable because the methodologies are generically approved for the WBN1 class of plants and because provision of appropriate analysis input values assures that the methodology applies specifically to WBN1.

The licensee also proposes to change BASES pages (B 3.2-2, B 3.2-4, B 3.2-13, B 3.2-14, B 3.5-2, B 3.5-3, B 3.5-4, B 3.5-12, and B 3.5-13) to reflect changes in the LOCA methodologies referenced, and differences in usage and results associated with the new references. These are acceptable because they are consistent with the added referenced LOCA methodologies as approved.

3.0 CLOSURE OF TRACKING NUMBERS

With the issuance of this safety evaluation, the staff has completed its activities for TAC Number MA6038 for the BE LBLOCA analysis. In a letter dated August 28, 1995, TVA provided commitments to re-perform the LBLOCA analysis. TVA provided a re-analysis with its initial (U)FSAR submittal on February 9, 1998, and has further provided an analysis based on the BE model as reviewed in this safety evaluation. Therefore, the staff's activities associated with TAC Number M89477 have now been completed.

TVA's letter of August 28, 1995 also provided a commitment to reperform the small break LOCA analysis. TVA provided information on the small break LOCA analysis with its letter of March 27, 1997, and with its initial (U)FSAR submittal on February 9, 1998. On the basis of this information, and the staff assessments included in Section 2.2 above, the staff's activities associated with TAC Number M93767 have been completed.

4.0 SUMMARY

The licensee has performed both LBLOCA and SBLOCA analyses with generically-approved methodologies. The licensee has shown that these methodologies apply to WBN1 by the plant's design and by indicating that TVA/W processes assure appropriate analysis input values. The licensee has performed new analyses of record implementing these methodologies and processes. The licensee will reflect new LOCA methodologies and the results of their analyses in WBN1 licensing documentation.

The NRC staff concludes that the methodologies and processes discussed in Section 2 are acceptable for licensing use at WBN1, and that these methodologies and processes are acceptable for inclusion in licensing documentation, including WBN1 (U)FSAR, TS, and the WBN1 COLR.

In Section 2.5, the NRC staff also concludes that the WBN1 TS changes proposed by the licensee in its June 25, 1999, submittal are acceptable.

5.0 STATE CONSULTATION

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

6.0 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 and changes surveillance requirements. 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 (65 FR 6411, dated February 9, 2000). 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.

7.0 CONCLUSION

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

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Date: March 17, 2000

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