September 11, 1997

Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: ISSUANCE OF AMENDMENT CYCLE 2 CORE RELOAD CHANGES (TAC NO. M98258)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment No. 7 to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant, Unit 1. This amendment is in response to your application dated March 27, as supplemented May 28 and June 4, 1997.

The amendment contains two parts which address Cycle 2 core design changes and provides operational enhancements for reactor trip setpoints. Part 1 addresses an increase in the containment sump boron concentration during a large break loss-of-coolant accident and describes changes to Technical Specification (TS) 3.5.1 and TS 3.5.4 regarding boron concentration. Part 2 addresses changes to TS Figure 2.1.1-1, TS Table 3.3.1-1, and TS 3.4.1 on safety limits, the trip system and pressure, temperature and flow limits, respectively. The supplement submitted on May 28, 1997 added a reference to Westinghouse topical report WCAP-12610-P-A for TS 5.9.5.b.

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

Sincerely,

/S/
Robert E. Martin, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

DF01 1/1

Docket No. 50-390

Enclosures: 1. Amendment No. 7 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

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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. 7
License No. NPF-90

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 27, as supplemented May 28, June 4 and July 30, 1997, 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 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. 7 , 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 no later than 30 days of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 11, 1997

ATTACHMENT TO AMENDMENT NO. 7

FACILITY OPERATING LICENSE NO. NPF-90

DOCKET NO. 50-390

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. * - Spill-over pages are provided to maintain document completeness.

Remove Pages	Insert Pages
2.0-2	2.0-2
3.3-18	3.3-18
3.3-21	3.3-21
3.3-22	3.3-22
3.3-37	3.3-37
3.4-1	3.4-1
3.4-2	3.4-2
3.5-2	3.5-2
3.5-10	3.5-10
5.0-32	5.0-32
B 2.0-4	B 2.0-4
B 2.0-5	B 2.0-5 *
B 2.0-6	B 2.0-6 *
B 3.2-13	B 3.2-13
B 3.3-17	B 3.3-17
B 3.3-18	B 3.3-18 *
B 3.3-19	B 3.3-19
B 3.3-24	B 3.3-24
B 3.3-25	B 3.3-25
B 3.3-28	B 3.3-28
B 3.3-29	B 3.3-29
B 3.3-30	B 3.3-30 *
B 3.3-31	B 3.3-31 *
B 3.3-32	B 3.3-32 *
B 3.3-33	B 3.3-33 *
B 3.3-34	B 3.3-34 *
B 3.3-35	B 3.3-35 *
B 3.3-92	B 3.3-92
B 3.4-2	B 3.4-2
B 3.5-26	B 3.5-26

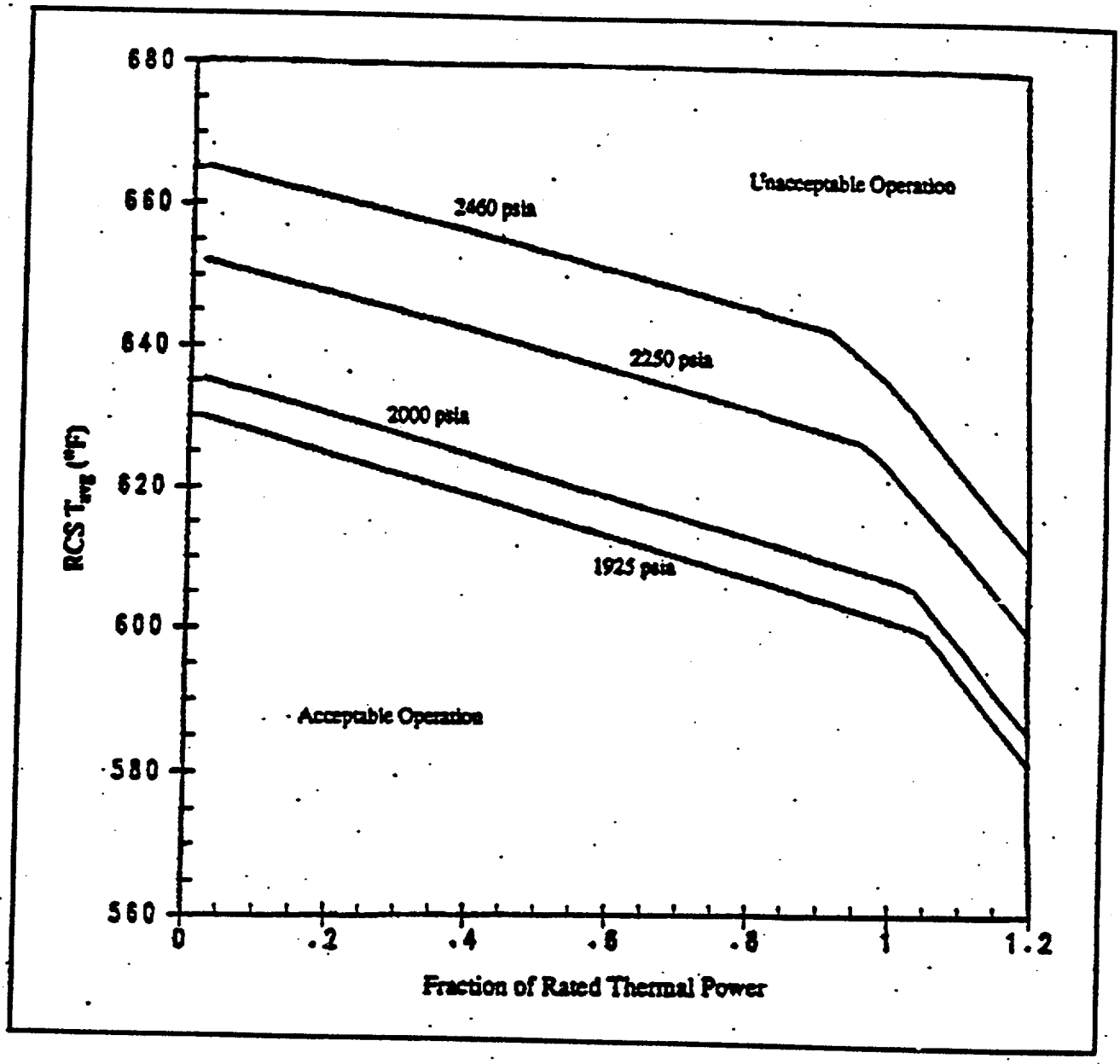

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

Table 3.3.1-1 (page 4 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
13. SG Water Level--Low-low Coincident with:	1,2	3/SG	U	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 16.4% of narrow range span	≥ 17% of narrow range span
a) Vessel ΔT Equivalent to power ≤ 50% RTP	1,2	3	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ 52.6% RTP	Vessel ΔT variable input ≤ 50% RTP
With a time delay (Ts) if one steam generator is affected					≤ 1.01 Ts (Refer to Note 3, Page 3.3-23)	≤ Ts (Refer to Note 3, Page 3.3-23)
or A time delay (Tm) if two or more Steam Generators are affected					≤ 1.01 Tm (Refer to Note 3, Page 3.3-23)	≤ Tm (Refer to Note 3, Page 3.3-23)
OR b) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2	3	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ 52.6% RTP	Vessel ΔT variable input ≤ 50% RTP
14. Turbine Trip						
a. Low Fluid Oil pressure	1(i)	3	O	SR 3.3.1.10 SR 3.3.1.14	≥ 43 psig	≥ 45 psig
b. Turbine Stop Valve Closure	1(i)	4	Y	SR 3.3.1.10 SR 3.3.1.14	≥ 1% open	≥ 1% open

(continued)

(i) Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 7 of 9)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.2% of ΔT span.

$$\Delta T \left\{ \frac{1+\tau_4 s}{1+\tau_5 s} \right\} \leq \Delta T_0 \left\{ K_1 - K_2 \frac{(1+\tau_1 s)}{(1+\tau_2 s)} [T - T'] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the indicated T_{avg} at RTP, ≤ 588.2°F.

P is the measured pressurizer pressure, psig
P' is the nominal RCS operating pressure, ≥ 2235 psig

$K_1 \leq 1.16$ $K_2 \geq 0.0183/°F$ $K_3 = 0.000900/psig$
$\tau_1 \geq 33$ sec $\tau_2 \leq 4$ sec
$\tau_4 \geq 3$ sec $\tau_5 \leq 3$ sec

$f_1(\Delta I) =$ $-2.62\{22 + (q_t - q_b)\}$ when $q_t - q_b < -22\%$ RTP
0 when -22% RTP $\leq q_t - q_b \leq 10\%$ RTP
$1.96\{(q_t - q_b) - 10\}$ when $q_t - q_b > 10\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

Table 3.3.1-1 (page 8 of 9)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.0% of ΔT span.

$$\Delta T\left[\frac{1+\tau_4 S}{1+\tau_5 S}\right] \le \Delta T_0 \left\{K_4 - K_5\left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6\left[T - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T'' is the indicated T_{avg} at RTP, ≤ 588.2°F.

$K_4 \le 1.10$ $K_5 \ge 0.02/°F$ for increasing T_{avg}; 0/°F for decreasing T_{avg} $K_6 \ge 0.00162/°F$ when $T > T''$; 0/°F when $T \le T''$

$\tau_3 \ge 5$ sec $\tau_4 \ge 3$ sec $\tau_5 \le 3$ sec

$f_2(\Delta I) = 0$ for all ΔI.

Table 3.3.2-1 (page 4 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. SG Water Level-Low Low	1,2,3	3 per SG	N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 16.4%	≥ 17.0%
Coincident with:						
1) Vessel ΔT equivalent to power ≤ 50% RTP	1,2	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ 52.6% RTP	Vessel ΔT variable input ≤ 50% RTP
With a time delay (Ts) if one S/G is affected					≤ 1.01 Ts (Note 1, Page 3.3-40)	≤ Ts (Note 1, Page 3.3-40)
or						
A time delay (Tm) if two or more S/G's are affected					≤ 1.01 Tm (Note 1, Page 3.3-40)	≤ Tm (Note 1, (Page 3.3-40)
OR						
2) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ 52.6% RTP	Vessel ΔT variable input ≤ 50% RTP

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2214 psig;

b. RCS average temperature ≤ 593.2°F; and

c. RCS total flow rate ≥ 380,000 gpm (process computer or control board indication).

APPLICABILITY: MODE 1.

----------NOTE----------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ≥ 2214 psig.	12 hours
SR 3.4.1.2	Verify RCS average temperature is ≤ 593.2°F.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 380,000 gpm (process computer or control board indication).	12 hours
SR 3.4.1.4	----------NOTE---------- Required to be performed within 24 hours after ≥ 90% RTP. ----------	
	Verify by precision heat balance that RCS total flow rate is ≥ 380,000 gpm.	18 months

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 ≥ 7717 gallons and ≤ 8004 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)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	----------NOTE---------- Only required to be performed when ambient air temperature is < 60°F or > 105°F. ---------- Verify RWST borated water temperature is ≥ 60°F and ≤ 105°F.	24 hours
SR 3.5.4.2	Verify RWST borated water volume is ≥ 370,000 gallons.	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ 2500 ppm and ≤ 2700 ppm.	7 days

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.

2. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

3. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 (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

SAFETY LIMITS (continued)

To meet the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. The effects of these uncertainties have been statistically combined with the correlation uncertainty to determine design limit DNBR values that satisfy the DNB design criterion. SL 2.1.1 reflects the use of the WRB-1 CHF correlation with design limit DNBR values of 1.25 and 1.24 for the typical and thimble cell, respectively.

Additional 10% DNBR margin is maintained by performing the safety analyses to a higher DNBR limit of 1.39 and 1.38 for the typical and thimble cell, respectively. This margin between the design and safety analysis limit is more than sufficient to offset known DNBR penalties (e.g., rod bow) and to provide the DNBR margin for operating and design flexibility.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

2.2.3

If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).

2.2.4

If SL 2.1.1 is violated, the Plant Manager, Site Vice President, and Nuclear Safety Review Board (NSRB) shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.5

If SL 2.1.1 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Reference 6). A copy of the report shall also be provided to the Plant Manager, Site Vice President, and NSRB.

2.2.6

If SL 2.1.1 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

2. Watts Bar FSAR, Section 7.2, "Reactor Trip System."

3. WCAP-8746-A, "Design Bases for the Overtemperature ΔT and the Overpower ΔT Trips," March 1977.

(continued)

BASES

REFERENCES
(continued)

4. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

5. Title 10, Code of Federal Regulations, Part 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors."

6. Title 10, Code of Federal Regulations, Part 50.73, "Licensee Event Report System."

7. WCAP-8762-P-A, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," July 1984.

8. Tong, L. S., "Boiling Crisis and Critical Heat Flux," AEC Critical Review Series, TID-25887, 1972.

9. Tong, L. S., "Critical Heat Fluxes on Rod Bundles," in "Two-Phase Flow and Heat Transfer in Rod Bundles," pages 31 through 41, American Society of Mechanical Engineers, New York, 1969.

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 large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;

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, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

- axial power distribution - the f(ΔI) Overtemperature ΔT Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation is included for delays associated with fluid transport from the core to the loop temperature detectors (RTDs), and thermowell and RTD response time delays.

ΔT_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured for each loop. T′ represents the 100% RTP Tavg value as measured by the plant for each loop. ΔT_0 and T′ normalize each loop's ΔT setpoint to the actual operating conditions existing at the time of measurement, thus forcing the setpoint to reflect the equivalent full power conditions as assumed in the accident analyses. Differences in RCS loop ΔT and Tavg can be due to several factors, e.g., measured RCS loop flow greater than minimum measured flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT and Tavg values. Loop specific values of ΔT_0 and T′ must be determined at the beginning fo each fuel cycle at full power, steady-state conditions (i.e., power distribution not affected by xenon transient conditions) and will be checked quarterly and adjusted, if required. Tolerances for ΔT_0 and T′ have been included in the determination of the Overtemperature ΔT setpoint.

The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature ΔT is indicated in two

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

loops. The pressure and temperature signals are used for other control functions. The actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux–High Setpoint trip.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and
- rate of change of reactor coolant average temperature—including dynamic compensation for delays associated with fluid transport from the core to the loop temperature detectors (RTDs), and thermowell and RTD response time delays.

ΔT_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured for each loop. T″ represents the 100% RTP Tavg value as measured by the plant for each loop. ΔT_0 and T″ normalize each loop's ΔT setpoint to the actual operating conditions existing at the time of measurement, thus forcing the setpoint to reflect the equivalent full power conditions as assumed in the accident analyses. Differences in RCS loop ΔT and Tavg can be due to several factors, e.g., measured RCS loop flow greater than minimum measured flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT and Tavg values. Loop specific values of ΔT_0 and T″ must be determined at the beginning fo each fuel cycle at full power, steady-state conditions (i.e., power distribution not affected by xenon transient conditions) and will be checked quarterly and adjusted, if required. Tolerances for ΔT_0 and T″ have been included in the determination of the Overpower ΔT setpoint.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Reactor Coolant Flow-Low (Single Loop) (continued)

the core. In MODE 1 below the P-8 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip (Function 10.b) because of the lower power level and the greater margin to the design limit DNBR.

The Reactor Coolant Flow-Low Trip Setpoint and Allowable Value are specified in % thermal design flow adjusted for uncertainties (95,000 gpm), however, the Eagle-21™ values entered through the MMI are specified in an equivalent % differential pressure.

b. Reactor Coolant Flow-Low (Two Loops)

The Reactor Coolant Flow–Low (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in two or more RCS loops while avoiding reactor trips due to normal variations in loop flow.

Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow–Low channels per loop to be OPERABLE.

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the Reactor Coolant Flow–Low (Two Loops) trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Reactor Coolant Flow-Low (Two Loops) (continued)

The Reactor Coolant Flow-Low Trip Setpoint and Allowable Value are specified in % thermal design flow adjusted for uncertainties (95,000 gpm), however, the Eagle-21™ values entered through the MMI are specified in an equivalent % differential pressure.

11. Undervoltage Reactor Coolant Pumps

The Undervoltage RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The voltage to each RCP is monitored. Above the P-7 setpoint, a loss of voltage detected on two or more RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Two Loops) Trip Setpoint is reached. The loss of voltage in two loops must be sustained for a length of time equal to or greater than that set in the time delay. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires one Undervoltage RCP channel per bus to be OPERABLE.

In MODE 1 above the P-7 setpoint, the Undervoltage RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

12. Underfrequency Reactor Coolant Pumps

The Underfrequency RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

13. Steam Generator Water Level-Low Low (continued)

(T_M) for the affected protection set, through the Man-Machine Interface. Failure of the vessel ΔT channel input (failure of more than one T_H RTD or failure of both T_C RTDs) affects the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the Man Machine Interface.

The LCO requires three channels of SG Water Level-Low Low per SG to be OPERABLE. This function initiates a reactor trip and the ESFAS function auxiliary feedwater pump start. The reactor trip feature is required to be OPERABLE in MODES 1 and 2 and the auxiliary feedwater pump start feature is required to be OPERABLE in MODES 1, 2, and 3.

In MODE 3, OPERABILITY of loop ΔT input to TTD is not required because MODE 3 ΔT = 0 (by definition). The Eagle-21™ code does not allow anything less than 0. The value of ΔT is low-limited to 0.0 prior to use in the calculation of the single and multiple trip time delays.

For MODES 1 and 2, ΔT_0, as used in the Vessel ΔT Equivalent to Power, represents the 100% RTP value as measured for each loop. ΔT_0 normalizes each loop's vessel ΔT to the actual operating conditions existing at the time of measurement, thus forcing the TTD to reflect the equivalent full power conditions as assumed in the accident analyses. Differences in RCS loop ΔT can be due to several factors, e.g., measured RCS loop flow greater than minimum measured flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Loop specific values of ΔT_0 must be determined at the beginning of each fuel cycle at full power, steady-state conditions (i.e., power distribution not affected by xenon transient conditions) and will be checked quarterly and adjusted, if required. Tolerances for ΔT_0 have been included in the determination of the Vessel ΔT Equivalent to Power.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

13. Steam Generator Water Level-Low Low (continued)

For MODES 1, 2, and 3, channel check surveillance testing on RCS loop ΔT input to TTD is not required. There are no provisions for performing a channel check on the RCS loop ΔT for the SG Level TTD Function. The power level can only be verified by connecting the Eagle-21™ Man-Machine Interface terminal and viewing the Dynamic Information for this channel. The Eagle-21™ system uses a redundant sensor algorithm for the hot leg and cold leg inputs, and will alert the operator if a failure occurs with the sensor or input signal conditioning.

The coefficients (A, B, C, D, E, F, G, and H) shown in the equation of Note 3 represent conservative values for the calculation of the time delay (i.e., the values given are 99% of the values used for the safety analyses). For the Eagle-21™ System, these coefficients are displayed (via the Man-Machine Interface) as A, B, C and D for the single request time delay, and E, F, G and H for the multiple request time delay.

In MODE 1 or 2, when the reactor is critical, the SG Water Level–Low Low trip must be OPERABLE. In MODES 1, 2, and 3 the normal source of water for the SGs is the Main Feedwater (MFW) System (not safety related). The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor in these MODES. The ESFAS Function of the SG Water Level-Low Low trip must be OPERABLE in MODES 1, 2, and 3. In MODES 3, 4, 5, and 6, the SG Water Level–Low Low trip Function does not have to be OPERABLE because the reactor is not operating or even critical.

14. Turbine Trip

a. Turbine Trip-Low Fluid Oil Pressure

The Turbine Trip–Low Fluid Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, approximately 50% power,

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Turbine Trip-Low Fluid Oil Pressure (continued)

will not actuate a reactor trip. Three pressure switches monitor the control oil pressure in the Turbine Electrohydraulic Control System. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function and RCS integrity is ensured by the pressurizer safety valves.

The LCO requires three channels of Turbine Trip-Low Fluid Oil Pressure to be OPERABLE in MODE 1 above P-9.

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the Turbine Trip-Low Fluid Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip-Turbine Stop Valve Closure

The Turbine Trip-Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level below the P-9 setpoint, approximately 50% power. This action will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function, and RCS integrity is ensured by the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Turbine Trip-Turbine Stop Valve Closure (continued)

pressurizer safety valves. This trip Function is diverse to the Turbine Trip-Low Fluid Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If all four limit switches indicate that the stop valves are all closed, a reactor trip is initiated.

The LSSS for this Function is set to assure channel trip occurs when the associated stop valve is completely closed.

The LCO requires four Turbine Trip-Turbine Stop Valve Closure channels, one per valve, to be OPERABLE in MODE 1 above P-9. All four channels must trip to cause reactor trip.

Below the P-9 setpoint, a load rejection can be accommodated by the Steam Dump System. In MODE 2, 3, 4, 5, or 6, there is no potential for a load rejection, and the Turbine Trip-Stop Valve Closure trip Function does not need to be OPERABLE.

15. Safety Injection Input from Engineered Safety Feature Actuation System

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. Reactor trip is not credited in the large break LOCA. However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Trip Setpoint and Allowable Values are not applicable to this Function. The SI Input is provided by solid state logic in the ESFAS. Therefore, there is no measurement signal with which to associate an LSSS.

The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

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APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

15. Safety Injection Input from Engineered Safety Feature Actuation System (continued)

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, and must be shut down in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE.

16. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE when the associated reactor trip Functions are outside the applicable MODES. These are:

a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel indicates approximately one decade above the minimum channel reading. If both channels decrease below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to increasing power above the source range; and

- on decreasing power, the P-6 interlock automatically enables the NIS Source Range Neutron Flux reactor trip.

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

16. Reactor Trip System Interlocks (continued)

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip may be blocked, and this Function would no longer be necessary. In MODE 3, 4, 5, or 6, the P-6 interlock is not required to be OPERABLE because the NIS Source Range is providing core protection.

b. Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either the Power Range Neutron Flux, P-10, or the Turbine Impulse Pressure, P-13 interlock. The LCO requirement for the P-7 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure—Low;
- Pressurizer Water Level—High;
- Reactor Coolant Flow—Low (Two Loops);
- Undervoltage RCPs; and
- Underfrequency RCPs.

These reactor trips are only required when operating above the P-7 setpoint (approximately 10% power). The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(2) on decreasing power, the P-7 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure—Low;
- Pressurizer Water Level—High;

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Low Power Reactor Trips Block, P-7 (continued)

- Reactor Coolant Flow–Low (Two Loops);
- Undervoltage RCPs; and
- Underfrequency RCPs.

Trip Setpoint and Allowable Value are not applicable to the P-7 interlock because it is a logic Function and thus has no parameter with which to associate an LSSS.

The P-7 interlock is a logic Function with train and not channel identity. Therefore, the LCO requires one channel per train of Low Power Reactor Trips Block, P-7 interlock to be OPERABLE in MODE 1.

The low power trips are blocked below the P-7 setpoint and unblocked above the P-7 setpoint.

In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the interlock performs its Function when power level drops below 10% power, which is in MODE 1.

c. Power Range Neutron Flux, P-8

The Power Range Neutron Flux, P-8 interlock is actuated at approximately 48% power as determined by two-out-of-four NIS power range detectors. Above approximately 48% power the P-8 interlock automatically enables the Reactor Coolant Flow–Low (Single Loop) reactor trip on low flow in one or more RCS loops on increasing power. The LCO requirement for this trip Function ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than approximately 48% power. On decreasing power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

c. Power Range Neutron Flux, P-8 (continued)

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9 interlock is actuated at approximately 50% power as determined by two-out-of-four NIS power range detectors. The LCO requirement for this Function ensures that the Turbine Trip–Low Fluid Oil Pressure and Turbine Trip–Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the combined capacity of the Steam Dump System and Rod Control System. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1.

In MODE 1, a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a load rejection beyond the capacity of the Steam Dump System.

e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at approximately 10% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 10% power on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Refer to the Bases for the Steam Generator Water Level Low-Low Reactor Trip, B 3.3.1, for a discussion of the required MODES and normalization of the vessel ΔT input to the TTD.

c. Auxiliary Feedwater-Safety Injection

An SI signal starts the motor driven and turbine driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.

d. Auxiliary Feedwater-Loss of Offsite Power

A loss of offsite power to the RCP buses will be accompanied by a loss of reactor coolant pumping power and the subsequent need for some method of decay heat removal. The loss of offsite power is detected by a voltage drop on each 6.9 kV shutdown board. Loss of power to either 6.9 kV shutdown board will start the turbine driven AFW pump to ensure that enough water is available to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.

Functions 6.a through 6.d (except the loop ΔT input to the trip time delay) must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. SG Water Level—Low Low in any operating SG will cause the motor driven AFW pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the SGs. SG Water Level—Low Low in any two operating SGs will cause the turbine driven pumps to start. These Functions do not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.7, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit of 2214 psig and the RCS average temperature limit of 593.2°F correspond to analytical limits of 2185 psig and 594.2°F used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO specifies limits on the monitored process variables—pressurizer pressure, RCS average temperature, and RCS total flow rate—to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error of 1.6% (process computer) or 1.8% (control board indication) based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to 1.7% (process computer) or 1.9% (control board indication).

Any fouling that might bias the flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations. The maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as 27 seconds, with offsite power available, or 37 seconds without offsite power.

For a large break LOCA analysis, the minimum water volume limit of 370,000 gallons and the lower boron concentration limit of 2500 ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 2700 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.7 TO FACILITY OPERATING LICENSE NO. NPF-90

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-390

1.0 INTRODUCTION

By letters dated March 27, May 28, June 4 and July 30, 1997, the Tennessee Valley Authority (the licensee or TVA) submitted a request for changes to the Watts Bar Nuclear Plant, Unit 1, Technical Specifications (TS). The requested amendment addresses changes in the Cycle 2 core design which will include a longer fuel cycle and more highly enriched fuel (from 3.1 percent to 3.7 percent). The refueling water storage tank (RWST) and accumulator boron concentrations will be increased to provide enough boron in the sump to meet the large break loss-of-coolant accident (LBLOCA) requirement for sump boron concentration. To mitigate temperature fluctuations and associated alarms, the overtemperature delta-temperature ($OT\Delta T$) and overpower delta-power ($OP\Delta T$) setpoints have been enhanced to increase the operating margin associated with these trip functions. In addition, the assumed plant reactor coolant system (RCS) flow is reduced from 97,500 gpm per loop to 93,100 gpm per loop (total of 390,000 gpm) to accommodate 10 percent steam generator tube plugging and a 2 percent reduction in thermal design flow. Also, a tolerance of 0.6°F will be implemented for the TS Surveillance for indicated differential temperature and a 1°F tolerance for the surveillance of T_{AVG} (identified as T prime and T double prime in the TS). Associated changes have been made to the TS Bases. The Core Operating Limits Report (COLR) methodologies listed in TS 5.9.5.b are amended to add a reference to the Westinghouse report WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report." The report reflects use of fuel assemblies in Cycle 2 using ZIRLO fuel rod cladding.

The July 30, 1997 submittal provided clarifying information which did not affect the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The amendment request contains two parts which address Cycle 2 core design changes and provides operational enhancements for reactor trip setpoints. Part 1 addresses an increase in the containment sump boron concentration during a large break loss of coolant accident (LOCA) and describes changes to TS 3.5.1, "Accumulator Boron Concentration," and TS 3.5.4, "RWST Boron Concentration." Part 2 addresses proposed changes to TS Figure 2.1.1-1,

ENCLOSURE

"Reactor Core Safety Limits," TS Table 3.3.1-1, "Reactor Trip System Instrumentation," and TS 3.4.1, "RCS Pressure, Temperature and Flow DNB [departure from nucleate boiling] Limits".

A supplement to the proposed amendment was submitted in a letter dated May 28, 1997 in which the addition of the reference to Westinghouse topical report WCAP-12610-P-A was proposed for TS 5.9.5.b.

The Cycle 2 core design for Watts Bar will include a longer fuel cycle and an increase in U-235 fuel enrichment from 3.1 weight percent (w/o) to 3.7 w/o. Both fresh fuel and spent fuel storage facilities at Watts Bar have been approved for storage of fuel assemblies of higher enrichment than 3.7 w/o U-235 in Amendment Number 6 to the facility operating license.

2.1 TS Figure 2.1.1-1, "Reactor Core Safety Limits"

Core safety limits, shown as the combination of thermal power, reactor coolant system (RCS) highest loop average temperature, and pressurizer pressure, in Figure 2.1.1-1, have been revised to account for the use of the Revised Thermal Design Procedure (RTDP) methodology for Cycle 2. The RTDP methodology, which has been previously approved by the staff, statistically combines uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions to obtain DNB design limits. Use of the RTDP methodology requires that variances in the plant operating parameters (pressurizer pressure, primary coolant temperature, reactor power, and reactor coolant system flow) be justified. Therefore, in support of the use of the RTDP, TVA submitted WCAP-14738, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Tennessee Valley Authority Watts Bar Unit 1," which addressed the changes to the instrument uncertainties for the primary system operating parameters. These uncertainty values are acceptable and they, or more conservative values, have been used in the RTDP analysis. Since the uncertainties in the initial operating conditions (i.e., pressure, temperature, power and flow) are accounted for in the calculation of the core design evaluation of the DNB safety analysis limit, they are not explicitly included in the transient analysis of DNB-related events which use the RTDP methodology. However, these uncertainties are appropriately applied to the initial operating conditions in other applicable transients which are not analyzed to investigate the minimum DNB response.

2.2 TS Table 3.3.1-1, "Reactor Trip System Instrumentation" Vessel ΔT Equivalent to Power Input to Steam Generator Water Level Low-Low Overtemperature ΔT and Overpower ΔT

The Watts Bar plant has experienced hot leg temperature fluctuations which tend to decrease the operating margin to both the $OT\Delta T$ and $OP\Delta T$ reactor trip setpoints. Although the staff does not consider these fluctuations to be a safety concern, they have impacted normal plant operation by causing OT alarms during steady-state operation. The licensee has proposed to modify the $OT\Delta T$ and $OP\Delta T$ setpoints and their allowable values and the allowable value for the vessel ΔT equivalent to power in order to decrease the sensitivity of these trip functions to the temperature fluctuations. These changes, as discussed

below, were based on the revised reactor core safety limits in TS Figure 2.1.1-1 discussed previously and are acceptable in that they provide adequate assurance that the safety limits will be met.

The specific changes to the OTΔT trip setpoint function include decreasing the allowable value from 2% to 1.2% of ΔT span, increasing K_1 from 1.0952 to 1.16, K_2 from .0133 to .0183/°F, K_3 from .000647 to .000900/psig, decreasing the value of τ_4 from 12 to 3 sec. The $f_1(\Delta I)$ function is changed from:

$$f_1(\Delta I) = \begin{cases} -1.34\{32 + (q_t - q_b)\} & \text{when } q_t - q_b < -32\% \text{ RTP} \\ 0 & \text{when } -32\% \text{ RTP} \le q_t - q_b \le 10\% \text{ RTP} \\ 1.22\{(q_t - q_b) - 10\} & \text{when } q_t - q_b > 10\% \text{ RTP} \end{cases}$$

to:

$$f_1(\Delta I) = \begin{cases} -2.62\{22 + (q_t - q_b)\} & \text{when } q_t - q_b) < -22\% \text{ RTP} \\ 0 & \text{when } -22\% \text{ RTP} \le q_t - q_b \le 10\% \text{ RTP} \\ 1.96\{(q_t - q_b) - 10\} & \text{when } q_t - q_b > 10\% \text{ RTP} \end{cases}$$

For the OPΔT function, the allowable value is decreased from 1.8% to 1.0%, K_4 is increased from 1.091 to 1.10, K_6 is increased from .00126 to .00162/°F, and τ_4 is decreased from 12 to 3 sec.

The vessel ΔT equivalent to power value changes from 52.7 % to 52.6 %.

The licensee will implement a tolerance of 0.6°F for the TS surveillance for ΔT_o and a 1°F tolerance for the surveillance of T_{AVG}. These tolerances have been incorporated as biases in the uncertainty analysis for the OTΔT, OPΔT, and vessel ΔT equivalent to power (used in the SG low-low water level reactor trip and engineered safety features actuation system (ESFAS) functions). To ensure consistency with the safety analysis, ΔT_o and T_{AVG} will be normalized quarterly.

2.3 <u>TS 3.4.1, "RCS Pressure, Temperature and Flow DNB Limits"</u>

The RCS average temperature limit has been reduced from 593.5°F to 593.2°F to account for the change in uncertainty from implementing the RTDP. The total RCS flow assumed in analyses has been reduced from 97,500 gpm per loop (total of 390,000 gpm) to 93,100 gpm per loop (total of 372,400 gpm) to accommodate 10% steam generator tube plugging and an additional 2% reduction in thermal design flow. The total flow value in the TS includes an allowance for instrument uncertainty. This TS value changes from an initial value of 397,000 gpm, as verified on the process computer or control boards, to 380,000 gpm. These values are consistent with the values used in the non-LOCA safety analyses for Cycle 2, which gave acceptable results, and their effects have been included in the revised core thermal limits of TS Figure 2.1-1. The changes are, therefore, acceptable.

2.4 <u>TS 3.5.1, "Accumulator Boron Concentration", TS 3.5.4, "RWST Boron Concentration", and TS basis B 3.5.4</u>

To accommodate the longer fuel cycle and higher fuel enrichment for Cycle 2 and future core designs, the RWST and accumulator boron concentrations will be increased to provide enough boron in the sump to ensure that the reactor core

will remain subcritical following a LBLOCA. Therefore, the following TS changes are proposed:

1) TS 3.5.1 - The boron concentration range from 1900 to 2100 parts per million (ppm) will be increased to a range of 2400 to 2700 ppm.
2) TS 3.5.4 - The boron concentration range from 2000 to 2100 ppm will be increased to a range of 2500 to 2700 ppm.
23) TS B 3.5.4 - The minimum RWST boron concentration will be increased from 2000 ppm to 2500 ppm to reflect the minimum value used in the post-LOCA sump analysis for core subcriticality. Also, the maximum RWST boron concentration will be increased from 2100 ppm to 2700 ppm to reflect the maximum value used in the calculation for the hot leg switch-over time.

The results of a calculation performed by Westinghouse indicated that the required sump boron concentration to maintain the core subcritical during a LBLOCA is 1741 ppm. This boron concentration is below the calculated sump concentration of 1855 ppm based on the minimum boron concentration specified in proposed TS 3.5.1 and TS 3.5.4. The licensee has also reduced the hot leg injection switch-over time from 12 hours to 9 hours into the event to prevent boron precipitation during post-LOCA long term recirculation.

The licensee has performed an evaluation of non-LOCA transients and accidents and has concluded that the proposed increase in the RWST and accumulator boron concentration would not adversely affect the consequences of any events. We have evaluated the licensee's submittal and conclude that the proposed TS changes are acceptable.

2.5 TS 5.9.5.b, "Core Operating Limits Report (COLR)"

The reference to WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report,", submitted by letter of N. J. Liparulo, Westinghouse, to NRC on April 10, 1995, is being added to include the methodology for specification 3.2.1, Heat Flux Hot Channel Factor, in the list of NRC-approved methodologies used to support the parameters specified in the COLR. This reference is being added because it reflects the methodology used for the rod heatup calculation in the LOCA evaluation model with consideration of ZIRLO clad fuel properties in support of the analysis for the heat flux hot channel factor. Fuel assemblies containing fuel rods clad with the advanced zirconium alloy material ZIRLO are being inserted into Cycle 2 and future Watts Bar cores. The use of ZIRLO is permitted by TS 4.2.1 and WCAP-12610-P-A is an NRC-approved methodology used for the rod heatup calculation in the LOCA evaluation model that is acceptable for use at Watts Bar. Accordingly, use of the report will ensure, in accordance with TS 5.9.5(c), that core operating limits will be determined such that all applicable limits, such as the emergency core cooling system limits, of the safety analysis are met. Therefore, the proposed addition is acceptable.

2.6 EVALUATION OF SETPOINT METHODOLOGY SUBMITTAL

The proposed amendment dated March 27, 1997 supports cycle 2 core design changes, revises reactor trip setpoints and changes to DNBR Limits including the incorporation of the Westinghouse Revised Thermal Design Procedure (RTDP)

Instrument Setpoint Methodology. TS regarding revisions to RCS DNB Limits, Overtemperature ΔT, Overpower ΔT, and the incorporation of the RTDP are reviewed here. Cycle 2 core design for Watts bar Unit 1 will include a longer fuel cycle and more highly enriched fuel. Additionally, WBN has experienced hot leg temperature fluctuations that decrease the operating margin to both the $OT\Delta T$ and $OP\Delta T$ reactor trip setpoints. TVA also proposes to reduce the plant thermal design flow to accommodate an assumption of up to 10% steam generator tube plugging. To obtain additional DNB margin for the $OT\Delta T$ and $OP\Delta T$ setpoints and reduced thermal design flow TVA also proposes to implement the RTDP (Westinghouse WCAP-14738, Revision 0).

This review is limited to instrumentation uncertainty methodology. The DNB analysis of the core incorporates the RTDP. The RTDP uncertainties are combined statistically to obtain the overall DNBR uncertainty factor such that the probability that DNB will not occur on the most limiting fuel rod is at least 95% (at a 95% confidence level) for any Condition I or II event. The above probability is based on the assumption that the uncertainties referenced can be represented with a random, normal, two sided probability distribution. This approach has been previously used by Westinghouse for numerous plants e.g., Wolf Creek, and others.

Instrumentation uncertainties are documented in the Watts Bar RTDP Instrument Uncertainty Methodology (WCAP-14738). Four operating parameter uncertainties are used in the uncertainty analysis of the RTDP. These parameters are pressurizer pressure, primary coolant temperature, reactor power, and reactor coolant system (RCS) flow. Reactor power is monitored by a secondary heat balance or calorimetric measurement every 24 hours. RCS flow is determined by the performance of a precision flow calorimetric after every refueling at the beginning of each fuel cycle. The RCS flow indicators are normalized to the precision calorimetric and used for monthly and daily surveillance. Pressurizer pressure is a control system parameter and the uncertainties associated with that system are included along with control board uncertainties. Similarly, T-average is also a controlled parameter and includes the control system uncertainties.

The RTDP combines error components for a channel by the square root sum of the squares (SRSS) for those uncertainty components found to be statistically independent. Errors that are determined to be dependent are combined arithmetically into independent groups and combined systematically. The described methodology is consistent with previous RTDP submittals, the Westinghouse setpoint methodology, and industry standards including ISA S67.04 1982 and Regulatory Guide (RG) 1.105, Revision 2, 1986, with respect to SRSS, the guidelines for combining various instrument uncertainties and the relationship between uncertainty components. WBN specific instrumentation data and procedures were reviewed and the uncertainty calculations completed based on the use of this data.

With the incorporation of the RTDP, the licensee has revised the DNB parameters for RCS T-Average, and RCS flow. The revision to TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits" is based on the incorporation of RTDP (WCAP-14738) which includes the use of a precision flow calorimetric at the beginning of each cycle to verify the TS

DNB reactor coolant system total flow rate limit and to normalize the RCS loop flow indicators used for the monthly and daily TS flow surveillance. The licensee will also revise the WBN Final Safety Analysis Report (FSAR) to reflect the incorporation of WCAP-14738.

Because the proposed modifications to the thermal design flow required a change to the core limits, the OTΔT and OPΔT setpoints were revised to reflect the core limit changes. Additionally, the licensee chose to revise the OTΔT and OPΔT setpoints to maximize operating margin (plant operating to trip setpoint). Further, tolerances were added for ΔTo, T' and T".

The Allowable Values for Vessel ΔT equivalent to power coincident input to Steam Generator Low-Low are also revised to reflect the additional tolerance assigned to the ΔT measurement.

WCAP-12096, "Westinghouse Setpoint Methodology for Protection Systems Watts Bar Unit 1 Eagle 21 Version," Revision 7, describes the protection system setpoint methodology used to combine the error components and determine instrumentation channel uncertainty terms specifically for Watts Bar. The methodology described in WCAP-12096 has been reviewed previously by the staff and found to be consistent with RG 1.105 Revision 2, 1986. WCAP-12096, Revision 7, has been revised to reflect the above setpoint changes for Watts Bar. The methodology, as updated in WCAP-12096, Revision 7, appropriately models the Watts Bar plant and is therefore acceptable. The references in the WBN FSAR and TS Bases to WCAP-12096 will be updated to reflect the current revision 7 of the WCAP.

Based on the above, the staff finds the methodology chosen by the licensee to be consistent with previously submitted RTDP and protection system setpoint methodologies, RG 1.105 Rev 2, and to be compatible with industry accepted standards including ISA-S67.04 1982, 1986. The proposed changes are, therefore, acceptable.

2.7 Summary

The staff has reviewed the proposed WBN TS changes submitted by TVA letters dated March 27, 1997, as supplemented May 28, June 4 and July 30, 1997 to support operation for cycle 2. As stated herein, the staff finds the proposed TS changes acceptable. The Bases have also been modified consistent with the revised TS requirements.

3.0 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.

4.0 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 (62 FR 35852, dated July 2, 1997). 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.

5.0 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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