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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT

DOCKET NO. 50-390

1.0 INTRODUCTION

By letters dated March 10, 1994; December 23, 1994; March 29, 1995; July 31, 1995; and September 8, 1995, Tennessee Valley Authority (TVA) submitted for staff review the setpoints for the cold overpressure mitigating system (COMS) and pressure-temperature (P-T) limits in the proposed "Watts Bar Unit 1 Pressure Temperature Limits Report" (PTLR), Revision 4. The submittal includes a plant-specific methodology report, "Heatup and Cooldown Limit Curves for Normal Operation for Watts Bar Unit 1," WCAP-13829, Revision 2. TVA's methodology is based on the Westinghouse topical report, "Methodology Used To Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," WCAP-14040, Revision 1, submitted by the Westinghouse Owners Group (WOG) on December 20, 1994, with an exception regarding the instrument uncertainties in the setpoints for COMS. WOG also sent additional information by letters dated June 16, July 18, and August 15, 1995.

In the late 1980s, the NRC, with input from the nuclear industry, initiated a program to streamline plant technical specifications (TS), thereby improving the overall safety of plant operation. One of the outcomes of the improvement program is the "Standard Technical Specifications (STS) For Westinghouse Plants," NUREG-1431, Revision 1, issued on April 7, 1995. The proposed Watts Bar PTLR is a part of an overall effort in converting the Watts Bar TS to the STS format. Under the STS format, the P-T limit curves and COMS setpoints may be removed from the Watts Bar TS and placed in a separate document, the "Pressure-Temperature Limits Report" (PTLR). Although relocated, the regulatory requirements for P-T and COMS limits are maintained in Limiting Condition for Operation (LCO) 3.4.3 and 3.4.12, respectively, in the Watts Bar TS. In addition, the PTLR will be administratively controlled in Section 5.9.6 of the Watts Bar TS. Presently, changes to the P-T and COMS limits in the plant TS require staff review and approval before implementation because of the license amendment process. With the PTLR concept, licensees may revise the P-T and COMS limits based on the 10 CFR 50.59 review process without the staff's prior approval.

The staff reviewed the proposed Watts Bar PTLR and methodology using the set of provisions (see Table 1) in a draft generic letter (GL) that was published in the *Federal Register* for public comment on June 2, 1995. The provisions are categorized into the following seven topics relevant to the P-T and COMS limits: (1) neutron fluence calculation, (2) reactor vessel material surveillance program, (3) low-temperature overpressure protection system, (4) adjusted reference temperature, (5) fracture mechanics calculation, (6) minimum temperature requirement, and (7) use of surveillance data.

ENCLOSURE

2.0 EVALUATION

2.1 Watts Bar Methodology

The Watts Bar P-T methodology is described in WCAP-13829 and WCAP-14040. The COMS setpoint methodology is described in WCAP-14040, Revision 1, except in the areas of (1) instrument uncertainties in the setpoints for COMS and (2) using the relief valves of the residual heat removal (RHR) system as a part of the COMS. In these two areas, the staff based its review on TVA's submittal dated September 8, 1995.

2.1.1 Provision 1: Neutron Fluence Calculation

The staff based its review on a Westinghouse-proposed expanded writeup regarding the neutron methodology in the additional information (Reference 1). The neutron fluence calculations are carried out using forward and adjoint formulations in r,θ geometry of the two-dimensional Discrete Ordinates Transport (DOT) code. The anisotropic scattering is treated with a P_3 expansion of the scattering cross section and the angular discretization is modeled with an S_8 order of angular quadrature. The core power distribution and the neutron source distribution were estimated conservatively, accounting for spectral changes due to plutonium accumulation. The revised version of the report uses the BUGLE-93 cross section library which is based on the data set of the Evaluated Nuclear Data File, version B-VI (ENDF/B-VI). The DOT code was rebenchmarked to the ENDF/B-VI cross sections using the Poolside Critical Assembly (PCA) simulator experiment at the Oak Ridge National Laboratory (ORNL), surveillance capsule, and cavity dosimetry measurements. It is stated that results of analytic sensitivity studies showed that the methodology is capable of providing best estimate fluence evaluations within ± 20 percent (1σ).

The methodology summarized above evolved over many years and has been validated using NRC-sponsored experimental measurements (i.e., the PCA experiment at ORNL as well as the reactor surveillance capsule data base). The latest improvement was the introduction of the ENDF/B-VI-based inelastic scattering cross sections for iron. In WCAP-14040, the DOT code was rebenchmarked with the new cross sections. Methodologies incorporating the same elements as in WCAP-14040 have been used in staff-sponsored work, and formed the basis for similar methodology approvals. The staff finds that WCAP-14040 incorporates state-of-the-art fast neutron radiation transport; therefore, the staff finds it acceptable.

2.1.2 Provision 2: Reactor Vessel Material Surveillance Program

The reactor vessel material surveillance program is designed to monitor radiation effects on reactor vessel materials under actual operating conditions. The radiation effects are determined from changes in fracture toughness of the material; this can be obtained by pre- and post-irradiation testing of vessel material specimens in surveillance capsules. Appendix H to 10 CFR Part 50 requires that the surveillance program satisfy ASTM Standard E-185 which specifies material testing, specimen sizes, specimen quantities, and material selection. The surveillance program at Watts Bar was not

mentioned in WCAP-14040 and WCAP-13829. However, the staff based its review on the Watts Bar surveillance program described in WCAP-9298, which is referenced in the Watts Bar PTLR. In addition, in a response to GL 92-01 dated July 7, 1992, TVA stated that the Watts Bar surveillance program was designed to satisfy the requirements of ASTM Standard E 185-73 (1973 Edition).

The Watts Bar surveillance program consisted of six surveillance capsules which are located in the reactor between the neutron shielding pads and the vessel wall. Each capsule contains 60 Charpy V-notch specimens, nine tensile specimens, twelve 1/2T compact tension specimens, and one bend bar specimen. The specimens were made from the intermediate shell forging 05 material, the weld metal, and the heat affected zone (HAZ) material. The specimens from intermediate shell forging 05 were machined in both the tangential orientation (long axis of the specimen parallel to the major working direction) and axial orientation (long axis of the specimen perpendicular to the major working direction). The weld specimens were machined so that the long axis was normal to the weld direction. The specific breakdown of the number of specimens for each material in each capsule is given in WCAP-9298.

The Charpy V-notch specimens will be tested at various temperatures to develop a complete transition curve in accordance with ASTM Standard E-23. The tensile specimens will be tested at room temperature, 300 °F, and 550 °F, in accordance with ASTM Standards E-8 and E-21. The 1/2T compact tension specimens will be tested dynamically in the ductile-to-brittle transition region and at upper-shelf initiation temperatures.

Each surveillance capsule also includes dosimeters and thermal monitors. The dosimeter wires are made of copper, iron, nickel, and aluminum with 0.15 weight percent cobalt. In addition, cadmium-shielded Np-237 and U-238 are included to measure the integrated flux at specific neutron energy levels for the individual capsule. The thermal monitors contain two low-melting-point eutectic alloys that would define the maximum temperature attained by the specimens during irradiation.

2.1.3 Provision 3: Low-Temperature Overpressure Protection System

The methodology for calculating setpoints for the low-temperature overpressure protection system, which is referred to as the cold overpressure mitigating system (COMS) by TVA, is discussed in Section 3.0 of WCAP-14040. COMS is designed to provide the capability, during reactor operation at low-temperature conditions, to automatically prevent the reactor coolant system (RCS) pressure from exceeding the applicable limits established by Appendix G to 10 CFR Part 50. COMS is manually enabled by reactor operators on the basis of its predetermined enable temperature during reactor startup and shutdown. WCAP-14040 specifies a methodology of developing a COMS which uses the power-operated relief valves (PORVs) with variable setpoints. After COMS is enabled, it will automatically function to mitigate overpressure. WCAP-14040 does not contain a methodology to use the relief valve(s) at the suction line of the RHR system in the design basis of COMS.

The design basis of COMS considers both mass-addition and heat-addition transients. WCAP-14040 defines the mass-addition transient as a mass

injection scenario when the RCS is water solid. The transient is postulated as the simultaneous isolation of the RHR and letdown systems coupled with a full charging pump flow because of the flow control failure. However, WCAP-14040 assumes only one charging pump running during the transient. The staff considers that the WCAP-14040 method is not conservative because it did not consider the worst-scenario case in which all charging pumps and safety injection pumps are running and injecting water into the RCS. The one-charging pump assumption is applicable only for plants whose TS restricts only one charging pump operable during the COMS operation. For those plants that do not have this TS restriction, an inadvertent actuation of safety injection may cause all operable charging and safety injection pumps to deliver flow to a water-solid RCS. In response to the staff's request for modification of its COMS design basis, WOG in its letter dated August 15, 1995, stated that Section 3.1 of WCAP-14040, Revision 1, will be modified to state the following: "Various combinations of charging and safety injection flows may also be evaluated on a plant-specific basis; however, the mass injection transient used as a design basis should encompass the limiting pump(s) operability configuration permitted per the plant-specific Technical Specification during the modes when COMS is required to be in operation." The staff considers that these additional design criteria would ensure that the most limiting mass-addition transients will be analyzed in the design of COMS.

For the heat-addition transient, WCAP-14040 assumes that the most limiting case is the startup of a reactor coolant pump (RCP) in a single loop with the RCS temperature being as much as 50 °F lower than the steam generator secondary-side temperature and the inadvertent isolation of the RHR system. This results in a sudden heat input to a water-solid RCS from the steam generators, creating an increasing pressure transient. This assumption is conservative provided that the plant TS restricts startup of an RCP when the steam generator secondary-side temperature is more than 50 °F higher than the RCS temperature.

The major function of COMS is to protect the structural integrity of the reactor vessel from excessive pressure and temperature loadings. In order to achieve this purpose, the P-T limits established for the RCS per the requirement of Appendix G to 10 CFR Part 50 are considered as the upper limits for the RCS during postulated transient conditions. However, since the overpressure events most likely occur during isothermal conditions in the RCS, the steady-state Appendix G limits are used for the design of COMS. Also, COMS provides for an operational consideration to maintain the integrity of the PORV piping. An upper pressure limit of 800 psia is selected for this purpose. This maximum pressure is selected on the basis of a generic study by Westinghouse using a type of PORV which would cause maximum back pressure in the piping during an overpressure transient. The lower limit of the RCS pressure during a transient is based on an operational consideration for maintaining a normal pressure differential across the RCP No. 1 seals for proper RCP operation. When there is insufficient range to protect both the upper (P-T limits) and lower (RCP seals) pressure limits, setpoints are selected to protect the P-T limits.

The methodology for developing the PORV setpoints intends to provide adequate protection for reactor vessel integrity and maintain proper

operational margins. In calculating the PORV setpoints, plant parameters and transient conditions listed in Section 3.2.1 of WCAP-14040 are considered. This list contains initial RCS and steam generator parameters, PORV size and lifting characteristics, mass and heat input rate to the RCS, pressure limits to be protected, and other parameters and conditions. These data were included in a specialized version of the LOFTRAN computer code which calculates the maximum and minimum RCS pressures due to overshoot and undershoot of the RCS pressure under various overpressure transient conditions. The function generator used to program the PORV setpoints curve has a number of programmable break points (typically nine points) in the code. The break points were selected so that the P-T limits are fully protected with the PORV setpoints curve established by connecting these break points. Each of the two PORVs may have a different pressure setpoints curve. The staggered setpoints for two PORVs would prevent excessive pressure undershoot that would challenge the RCP No. 1 seal performance criteria. However, each PORV with its setpoints will protect P-T limits, assuming a single failure of the other PORV.

Section 3.2 of WCAP-14040 indicates that since the P-T limits are conservatively determined, the uncertainties in the pressure and temperature instrumentation utilized by COMS are not explicitly accounted for in the selection of the PORV setpoints for COMS. The staff finds this approach unacceptable. The staff requires that WCAP-14040, Revision 1 be modified to incorporate instrument uncertainties into the PORV setpoints for COMS to ensure that the P-T limits are protected. In a letter dated September 8, 1995, TVA supplemented its COMS methodology to include instrument uncertainties at the Watts Bar plant. TVA has committed to incorporate instrument uncertainties in its determination of PORV setpoints for COMS at Watts Bar. The methodology for determining instrument uncertainties conforms to the Instrument of America (ISA) Standard, ISA-67.04, which is endorsed by the NRC in Regulatory Guide (RG) 1.105.

The "enable" temperature is the RCS temperature below which COMS is required to function. This temperature is specified in Branch Technical Position RSB 5-2, attached to Standard Review Plan (SRP) Section 5.2.2. It is the RCS water temperature corresponding to a metal temperature of at least RT_{NDT} + 90 °F at the 1/4t or 3/4t reactor vessel beltline location (t = beltline thickness). Above this temperature, brittle fracture of the reactor vessel is not expected. The enable temperature calculation in WCAP-14040 is consistent with the staff position stated in Branch Technical Position RSB 5-2 and, therefore, is acceptable. The staff concludes that the COMS methodology is reasonable and acceptable.

WCAP-14040 discussed ASME Code Case N-514, which provides an alternative setpoint method to Appendix G to Section XI of the ASME Code and SRP Section 5.2.2. The code case allows (1) the maximum pressure of the PORV setpoints to 110 percent of the P-T limits, and (2) the enable temperature of 200 °F or the RCS temperature corresponding to a reactor vessel metal temperature of at least RT_{NDT} + 50 °F, whichever is greater. Code Case N-514, which is incorporated into Appendix G to Section XI of the ASME Code (1994 Addenda), has not been endorsed by NRC regulations. As a reactor vessel ages, embrittlement increases and the plant's pressure-temperature operating window at low temperature decreases. When Code Case N-514 is approved, it increases the size of the operating window and allows additional flexibility in plant operation to decrease the potential for undesirable lifting of the PORVs. Since Watts Bar has not been operated, embrittlement has not yet decreased its operating window. Hence, Code Case N-514 is not needed at this time.

2.1.4 Provision 4: Adjusted Reference Temperature (ART)

TVA's methodology for determining the limiting ART conforms to RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The ART is calculated by adding the initial nil-ductility transition reference temperature of the unirradiated material (IRT_{NDT}), the shift in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin to account for uncertainties in the prediction method as follows:

$$ART = IRT_{NDT} + \Delta RT_{NDT} + margin \qquad (1)$$

WCAP-14040 provides guidance on the derivation of the IRT_{NDT} as defined in paragraph NB-2331 of Section III of the ASME Code and NRC Branch Technical Position MTEB 5-2. The value is derived from results of a series of Charpy V-notch impact tests and drop-weight tests. Initially, a temperature, T_{NDT}, at or above the nil-ductility transition temperature, is determined by drop-weight tests. Next, at a temperature not greater than T_{NDT} + 60 °F, each specimen of the Charpy V-notch test shall exhibit at least 35 mils of lateral expansion and at least 50 ft-lb of absorbed energy. If the two requirements are met, T_{NDT} is the IRT_{NDT}. If the two requirements are not met, additional Charpy V-notch tests (in groups of three specimens) are performed to determine the temperature, T_{CV}, at which the requirements are met. In this case, T_{CV} - 60 °F is the IRT_{NDT}. If the Charpy V-notch test has not been performed at T_{NDT} + 60 °F or, if the test at this temperature does not exhibit the two requirements, the IRT_{NDT} can be obtained by a full Charpy impact curve developed from the minimum data points of all the Charpy tests performed.

The calculation of the mean value of ΔRT_{NDT} due to irradiation conforms to RG 1.99, Revision 2 as follows:

$$\Delta RT_{NDT} = CF \times f^{(0.28-0.10\log f)} \qquad (2)$$

where, CF is the chemistry factor, and f is the fast neutron fluence at a specific depth. The chemistry factor is calculated by using either copper and nickel contents, or credible plant-specific surveillance data. The fast neutron fluence is calculated for any depth by the following equation:

$$f = f_{surface} \times \exp(-0.24*) \qquad (3)$$

where, $f_{surface}$ is the neutron fluence at the base metal surface of the vessel at the location of the postulated defect and * (in inches) is the depth into the vessel wall measured from the interface of vessel cladding and base metal.

The margin is included in the ART calculations to obtain a conservative, upper-bound value of ART for the calculation required by Appendix G to 10 CFR Part 50. The margin is calculated by the following equation:

$$Margin = 2\sqrt{(\sigma_i^2 + \sigma_\Delta^2)} \qquad (4)$$

where, σ_i is the standard deviation for IRT_{NDT} and σ_Δ is the standard deviation

for ΔRT_{NDT}. If IRT_{NDT} is a measured value, σ_i is estimated from the precision of the test method. For generic mean values, σ_i is the standard deviation from the set of data used to establish the mean. σ_Δ is 28 °F for welds and 17 °F for base metal per RG 1.99, Revision 2. σ_Δ is reduced by half, when surveillance data are used. σ_Δ need not exceed half the mean value of ΔRT_{NDT} for all cases.

2.1.5 Provision 5: Fracture Mechanics Calculation

TVA used linear elastic fracture mechanics in Appendix G to Section XI of the ASME Code in calculating allowable pressure-temperature limits. The method is based on restricting the stress intensity factor of the postulated defect to be less than the reference stress intensity factor of the reactor vessel material, K_{Ia}. The K_{Ia} is determined by the metal temperature at the tip of the postulated flaw and the RT_{NDT} at the same location. The flaw is assumed to have a depth of one-fourth of the section thickness and a length of 1.5 times the section thickness. The K_{Ia} curve in the ASME Code is given by the following equation:

$$K_{Ia} = 26.78 + 1.223 \times \exp[0.0145(T-RT_{NDT} + 160)] \qquad (5)$$

where, T is the metal temperature and RT_{NDT} is the metal reference nil-ductility transition temperature (ART value) of the limiting vessel material at the 1/4t and 3/4t locations of the vessel wall. In Appendix G to Section III of the ASME Code (1995 Edition), the reference stress intensity factor is denoted as K_{IR}, whereas, in Appendix G to Section XI, the reference stress intensity factor is denoted as K_{Ia}. However, K_{IR} and K_{Ia} curves are identical and their equations have the same functional forms and coefficients.

The stress intensity factor caused by the postulated crack is limited to the reference stress intensity factor of the vessel material as follows:

$$C \times K_{IM} + K_{IT} < K_{Ia} \qquad (6)$$

where, K_{IM} is the stress intensity factor caused by pressure (membrane) stress, K_{IT} is the stress intensity factor caused by the thermal gradients through the vessel wall, and C is a safety factor that is 2 for heatup and cooldown and 1.5 for hydrostatic and leak test conditions when the reactor core is not critical.

The K_{IT} is determined using the one-dimensional heat conduction equation and boundary conditions:

$$\rho C \, dT/dt = K \, [d^2T/dr^2 + 1/r \, dT/dr] \qquad (7)$$

$$\text{at } r = r_i, \; -K \, dT/dr = h(T-T_C) \qquad (8)$$

$$\text{at } r = r_o, \quad dT/dr = 0 \qquad (9)$$

where, r_i and r_o are the reactor vessel inner and outer radius, respectively; ρ is the material density; C is the material specific heat; K is the material thermal conductivity; T is the local temperature; r is the radial location; t

is the time; h is the heat transfer coefficient between the coolant and the vessel wall; and T_c is the coolant temperature. Solving Equations 7, 8, and 9 gives a vessel-wall location and time-dependent temperature distribution for all heatup and cooldown rates. This temperature distribution is placed in the following equation for thermal stress in a hollow cylinder:

$$\sigma_\theta = \frac{E\alpha}{(1-v)} \frac{1}{r^2} \left[\frac{(r^2 + r_i^2)}{(r_o^2 - r_i^2)} \int_{r_i}^{r_o} T(r,t)\ rdr + \int_{r_i}^{r} T(r,t)\ rdr - T(r,t)\ r^2 \right]$$

where, E is the modulus of elasticity, α is the coefficient of linear expansion, and v is Poisson's ratio. Solving this equation yields a position and time-dependent distribution of hoop thermal stress, $\sigma_\theta(r,t)$. The linear bending (σ_b) and constant membrane (σ_m) stress of the hoop thermal stress are approximated by the linearization technique in Appendix A of Section XI of the ASME Code. After determining the bending and membrane stresses, the K_{IT} is calculated by the following equation:

$$K_{IT} = [1.1\ \sigma_m\ M_k + \sigma_b\ M_b]\ \sqrt{(\pi a/Q)} \qquad (10)$$

where, M_k and M_b are correction factors for membrane and bending stresses, respectively, and Q is the flaw shape factor. TVA's use of Equation 10 is consistent with the Welding Research Council Bulletin 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials." The K_{IT} solution will be used in combination with K_{Ia} (Equation 5) to solve for $K_{IM(max)}$ using Equation 6:

$$K_{IM(max)} = (K_{Ia} - K_{IT})/2.0 \qquad (11)$$

The maximum allowable pressure stress for a given temperature is determined using an iterative process and the following three equations:

$$Q = \phi^2 - 0.212\ (\sigma_p/\sigma_y)^2 \qquad (12)$$

$$\sigma_p = K_{IM(max)}/[1.1\ M_k\ \sqrt{(\pi a/Q)}] \qquad (13)$$

$$K_{IP} = 1.1\ M_k\ \sigma_p\ \sqrt{(\pi a/Q)} \qquad (14)$$

where ϕ is the elliptical integral of the second kind, σ_p is the pressure stress, σ_y is the yield stress, M_k is the correction factor for constant membrane stress, a is the crack depth at the 1/4t location, and K_{IP} is the stress intensity factor caused by vessel internal pressure.

The solution to the iterative process, σ_p, is used to calculate the maximum allowable internal pressure as a function of temperature as given in the following equation:

$$P(T_c) = \sigma_p\ [(r_o^2 - r_i^2)/(r_o^2 + r_i^2)] \qquad (15)$$

The steady-state, cooldown, and heatup P-T curves can be determined using this process. For steady state, K_{IT} is zero and K_{Ia} is determined at the 1/4t location (the most restrictive location). For cooldown, K_{IT} and K_{Ia} are determined at the 1/4t location. The pressure-temperature curve at 1/4t is

compared with the steady-state curve. The allowable pressure for cooldown is determined by the lesser of the two values, and the resulting curve is the composite cooldown limit curve. For heatup, K_{IT} and K_{Ia} are determined at the 1/4t and 3/4t locations. The P-T curves at 1/4t, 3/4t, and steady state are compared. The lowest of the three for each heatup rate is used to generate the composite heatup limit curve. The composite cooldown limit curve and composite heatup limit curve provide the allowable operating range for operation. The staff finds the WCAP-14040 methodology consistent with Appendix G to Section III of the ASME Code and SRP Section 5.3.2.

2.1.6 Provision 6: Minimum Temperature Requirement

Appendix G to 10 CFR Part 50 imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. When the core is critical (other than for the purpose of low-level physics test), the temperature of the reactor vessel must not be lower than 40 °F above the minimum temperature of heatup and cooldown curves and must not be lower than the minimum temperature for the inservice hydrostatic pressure test. TVA adopted these requirements in its methodology and imposed the restrictions on its P-T limit curves.

2.1.7 Provision 7: Use of Surveillance Data

WCAP-14040 stated that when two or more credible surveillance capsules have been removed, the measured increase in ΔRT_{NDT} must be compared to the predicted increase in RT_{NDT} for each surveillance material. The predicted increase in RT_{NDT} is the mean shift in RT_{NDT} calculated by Equation (2) plus two standard deviations specified in RG 1.99, Revision 2. If the measured value exceeds the predicted value ($\Delta RT_{NDT} + 2\sigma_A$), a supplement to the PTLR must be provided to demonstrate how the results affect the approved methodology.

2.2 Watts Bar PTLR

2.2.1 Provision 1: Neutron Fluence Calculation

TVA provided maximum fluence at the inside surface of the vessel as 6.96E18, 3.18E19, and 4.77E19 neutron/cm^2 at 7, 32, and 48 effective full power years (EFPY), respectively, in the PTLR. The maximum fluence is located at the 25° azimuthal angle. TVA conservatively applied the maximum fluence to all beltline materials even though some beltline materials are not located at the 25° azimuthal angle when calculating the ART. The staff compared the neutron fluence at 32 EFPY in the PTLR to the neutron fluence submitted under GL 92-01 and found that the neutron fluences are consistent. The neutron fluence will be updated when the test results of dosimeters removed from surveillance capsules become available.

2.2.2 Provision 2: Reactor Vessel Material Surveillance Program

The Watts Bar PTLR contains the surveillance capsule removal schedule for all six capsules in the surveillance program. A capsule will be withdrawn at the first refueling outage, by 5.4 EFPY, by 8.9 EFPY, and by 17.8 EFPY. Two capsules will be held in standby. After the second capsule is removed, TVA will review the results of the capsule analyses and may revise the capsule removal schedule accordingly. TVA may use the two standby capsules for additional monitoring. In addition, the surveillance data (if credible) can be used to revise the ART as directed by RG 1.99, Revision 2.

The surveillance capsule withdrawal schedule provides adequate assurance for monitoring of the radiation effects on the reactor vessel materials. The staff verified that the applicant's surveillance program is in compliance with Appendix H to 10 CFR Part 50.

2.2.3 Provision 3: Low-Temperature Overpressure Protection System

The methodology for the low-temperature overpressure protection system, also known as COMS, is described in Section 3.0 of Watts Bar PTLR, Revision 4. In addition, TVA submitted the treatment of COMS instrument uncertainties in a letter dated September 8, 1995. The PORV setpoints for COMS were developed to protect the 1.5 EFPY P-T limit curves. The calculated PORV setpoints are specified in Figures 3.1-1 through 3.1-4 and Table 3.1-1 of the PTLR. The P-T limits for the design of COMS are contained in the 1.5 EFPY curves for heatup and cooldown which are specified in Figures 3.1-5 and 3.1-6, and Tables 3.1-2 and 3.1-3. These setpoint values are not compensated for pressure difference between the pressure transmitter and the reactor midplane/beltline or for instrument inaccuracies. Section 3.1 of the PTLR states that these calculated setpoints are compensated for pressure differential and instrument inaccuracies in the Site Engineering Setpoint and Scaling (SESS) documents for instrument loop numbers 1-T-68-1B and 1-T-68-43B to obtain the PORV lift limits which are the actual PORV setpoints to be applied for COMS. By letter dated September 8, 1995, in a response to the staff request, TVA provided data from the SESS documents to demonstrate the adequacy of the compensation process regarding pressure differential and instrument inaccuracies. Considering the SESS documents as an extension of the PTLR, the staff finds that the Watts Bar PTLR has adequately determined the PORV setpoints in accordance with the Watts Bar plant's specific methodology for COMS.

The proposed Watts Bar TS 3.4.12 allows the use of the RHR suction relief valve with a setpoint of 436.5 psig as an alternative to one PORV for COMS. In response to the staff question on the adequacy of the RHR relief valve to perform the design function of COMS, TVA in its letter dated September 8, 1995, reported the results of an analysis which demonstrate that the RHR relief valve with a setpoint of 436.5 psig will protect the P-T limits established for Watts Bar under the most limiting overpressure transient conditions. The limiting mass-addition transient assumed for Watts Bar COMS is one charging pump injecting water into the RCS at 470 gpm, which is less than the RHR relief valve capacity of 480 gpm. The licensee has assumed a 3 percent drift of the setpoint for the RHR relief valve. The results of the licensee's analysis indicated that the RHR valve can protect P-T limits with

sufficient margin. The staff finds the use of the RHR relief valve as an alternative to PORV for COMS acceptable.

2.2.4 Provision 4: Adjusted Reference Temperature (ART)

The ART value is a measure of the embrittlement of the reactor vessel materials caused by neutron irradiation. The ART should be calculated in accordance with RG 1.99, Rev. 2 as specified by GL 88-11. RG 1.99 defines the ART as the sum of initial nil-ductility transition reference temperature (IRT_{ndt}) of the reactor vessel material, the increase in RT_{ndt} caused by neutron irradiation (ΔRT_{NDT}), and a margin to account for uncertainties in the prediction method. The ΔRT_{NDT} is calculated from the product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the vessel material.

The Watts Bar reactor beltline was fabricated with intermediate shell forging 05, lower shell forging 04, and circumferential weld, heat No. 895075. Both forgings were made from a nickel-chromium-molybdenum steel, SA 508 Class 2. TVA calculated an ART for the limiting material, intermediate shell forging 05, of 181.1 °F at the 1/4t location (t is the thickness at beltline of the vessel) and 147.7 °F at the 3/4t location for 7 EFPY based on Position C.1 in RG 1.99. Forging 05 has an IRT_{ndt} of 47 °F and a copper and nickel content of 0.17 percent and 0.80 percent, respectively. The staff confirmed the copper and nickel contents and IRT_{NDT} of forging 05 with respect to the data from TVA's responses to GL 92-01 dated July 7, 1992; October 15, 1993; and June 13, 1994. The staff also performed an independent calculation of the ART and verified that TVA's limiting ARTs are acceptable.

TVA also calculated pressurized thermal shock value, RT_{PTS}, for the two reactor vessel forgings and circumferential weld based on the method described in 10 CFR 50.61. TVA calculated a limiting RT_{PTS} of 253 °F for intermediate shell forging 05. On the basis of its calculation, the staff confirmed that the limiting RT_{PTS} value is correct and is within the screening criteria of 10 CFR 50.61.

2.2.5 Provision 5: Fracture Mechanics Calculation

The staff evaluated the Watts Bar P-T limits based on Appendix G to 10 CFR Part 50 and SRP Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P-T limits for the reactor vessel must be at least as conservative as those obtained by Appendix G to Section III of the ASME Code.

SRP Section 5.3.2 provides guidance on calculating the P-T limits using linear elastic fracture mechanics methodology specified in Appendix G to Section III of the ASME Code. The linear elastic fracture mechanics methodology postulates sharp surface defects that are normal to the direction of maximum stress and have a depth of one-fourth of the section thickness (1/4t) and a length of 1-1/2 times the section thickness of the beltline region of the reactor vessel. The critical locations in the vessel for this methodology are the 1/4t and 3/4t locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

Substituting the calculated ARTs into equations in SRP Section 5.3.2, the staff verified that the proposed 7 EFPY P-T limits for heatup, cooldown, criticality, and inservice hydrostatic test in the Watts Bar PTLR satisfy the requirements in paragraphs IV.A.2 and IV.A.3 of Appendix G to 10 CFR Part 50.

The Watts Bar methodology in WCAP-13829, Revision 2, also included heatup and cooldown curves for 7 EFPY based on ASME Code Case N-514 (Figures 3 and 4). Because the NRC has not endorsed the code case, TVA indicated that the 7 EFPY curves in WCAP-13829 will not be used. The 7 EFPY curves in the Watts Bar PTLR which were calculated without Code Case N-514 will be used.

2.2.6 Provision 6: Minimum Temperature Requirement

In addition to beltline materials, Appendix G to 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. On the basis of the initial RT_{ndt} of -42 °F of the closure head flange (provided by TVA), the staff has determined that the proposed P-T limits comply with the requirement for the closure flange region during normal operation, hydrostatic pressure test, and leak test.

2.2.7 Provision 7: Use of Surveillance Data

The Watts Bar plant has not been operated; therefore, surveillance data are not available. TVA has stated that, when the surveillance data are available, they will be placed in the Watts Bar PTLR.

3.0 CONCLUSION

The proposed Watts Bar PTLR and associated methodology are acceptable because the staff has determined the following:

(1) The neutron fluence calculation conforms to the state-of-the-art fast neutron radiation transport methodology.

(2) The reactor vessel surveillance program conforms to the requirements of Appendix H to 10 CFR Part 50.

(3) The proposed 7 EFPY P-T limits for heatup, cooldown, inservice hydrostatic test, and criticality in the Watts Bar PTLR conforms to the requirements of Appendix G to 10 CFR Part 50, SRP Section 5.3.2, and GL 88-11.

(4) The material properties and chemistry used in calculating the P-T limits are consistent with data submitted under GL 92-01.

(5) The proposed COMS setpoints are valid for 1.5 EFPY because they conform to SRP Section 5.2.2 and Branch Technical Position RSB 5-2.

(6) The contents in the Watts Bar PTLR and methodology report conform to the staff requirements as described in a draft generic letter published in the *Federal Register* for public comment on June 2, 1995.

4.0 REFERENCE

1. Letter from L. Bush (Westinghouse Owners Group) to USNRC Document Control Desk (Attention Chief, Planning, Program and Management Support Branch), "Response to Concerns Identified During Review of WCAP-14040, Revision 1, 'Methodology Used To Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,'" June 16, 1995.

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TABLE 1

REQUIREMENTS FOR METHODOLOGY AND PTLR

PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN STS	MINIMUM REQUIREMENTS TO BE INCLUDED IN METHODOLOGY	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR
1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).	Describe transport calculation methods, including computer codes and formulas used to calculate neutron fluence. Provide references.	Provide the values of neutron fluences that are used in the adjusted reference temperature (ART) calculation.
2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR Part 50. The withdrawal schedule for reactor vessel material surveillance specimens shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.	Briefly describe the surveillance program. Licensee transmittal letter should identify by title and number the report containing the Reactor Vessel Surveillance Program and surveillance capsule reports. Topical/generic report contains placeholder only. Reference Appendix H to 10 CFR Part 50.	Provide the surveillance capsule withdrawal schedule, or reference by title and number those documents that contain the schedule. Reference the surveillance capsule reports by title and number if ARTs are calculated using surveillance data.
3. Low temperature overpressure protection (LTOP) system lift setting limits developed using NRC-approved methodologies may be included in the PTLR.	Describe how the LTOP system limits are calculated applying system/thermal hydraulics and fracture mechanics. Reference SRP Section 5.2.2; Code Case N-514; ASME Code, Appendix G, Section XI as applied in accordance with 10 CFR 50.55.	Provide setpoint curves or setpoint values.
4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for irradiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.	Describe the method for calculating the ART using Regulatory Guide 1.99, Revision 2.	Identify both the limiting ART values and limiting materials at the 1/4t and 3/4t locations (t = vessel beltline thickness). PWRs - identify RT_{PTS} value in accordance with 10 CFR 50.61

REQUIREMENTS FOR METHODOLOGY AND PTLR (Continued)

PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN STS	MINIMUM REQUIREMENTS TO BE INCLUDED IN METHODOLOGY	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR
5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800, SRP Section 5.3.2, "Pressure-Temperature Limits."	Describe the application of fracture mechanics in constructing P-T curves based on ASME Code, Appendix G, Section XI, and SRP Section 5.3.2.	Provide the P-T curves for heatup, cooldown, criticality, and hydrostatic and inservice leak tests.
6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.	Describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P-T curves.	Identify minimum temperatures on the P-T curves,such as minimum boltup temperature and hydrotest temperature.
7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}, where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.	Describe procedure if measured value exceeds predicted value. <u>WHEN OTHER PLANT DATA ARE USED</u> 1. Identify the source(s) of data. 2.a Identify by title and number the safety evaluation report that approved the use of data for the plant. Justify applicability. **OR** 2.b Compare licensee data with other plant data for both the radiation environments (e.g., neutron spectrum, irradiation temperature) and the surveillance test results.	Provide supplemental data and calculations of the chemistry factor in the PTLR if the surveillance data are used in the ART calculation. Evaluate the surveillance data to determine if they meet the credibility criteria in Regulatory Guide 1.99, Revision 2. Provide the results.