TVA

Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381

July 17, 1992

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Gentlemen:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

WATTS BAR NUCLEAR PLANT (WBN) - PROPOSED CHANGES TO FINAL SAFETY ANALYSIS REPORT (FSAR) RESULTING FROM REANALYSIS OF PRESSURIZED THERMAL SHOCK (TAC M77896)

TVA has recently completed reanalyzing the reactor vessels of both WBN units for their susceptibility to pressurized thermal shock (PTS). The analysis demonstrated WBN's compliance with 10 CFR 50.61 and its fracture toughness requirements for protection against PTS events. The analysis used the calculational methodology developed by Westinghouse and incorporated updated input parameters for WBN's reactor vessels and core components. It also addressed a concern about the peak neutron fluence that could be experienced by the reactor vessel of each WBN unit. This concern was raised by Mr. Lambros Lois and Mr. Peter Tam of the NRC staff in a telephone conversation on November 18, 1991.

Preliminary FSAR page markups reflecting the latest PTS analysis are attached to this letter as Enclosure 1. Enclosures 2 and 3 provide the technical bases for these FSAR changes. Enclosure 2 is Westinghouse Topical Report WCAP-13300 ("Evaluation of Pressurized Thermal Shock for Watts Bar Unit 1"). Enclosure 3 is WCAP-13301 ("Evaluation of Pressurized Thermal Shock for Watts Bar Unit 2").

Please review the enclosed set of proposed changes to FSAR Chapter 5. TVA plans to incorporate the changes formally in a future FSAR amendment after receiving NRC concurrence.

A001
~~A053~~ 1/1

If you have any questions, please telephone John Vorees at (615) 365-8819.

Sincerely,

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ENCLOSURE 1

(FSAR Page Markups)

Material specifications used for the principal pressure retaining applications in each component comprising the Reactor Coolant Pressure Boundary (RCPB) are listed in Table 5.2-8 for ASME Class 1 Primary Components and Table 5.2-9 for ASME Class 1 and 2 Auxiliary Components. The materials are procured in accordance with the material specification requirements and include the special requirements of the ASME Boiler and Pressure Vessel Code, Section III, plus Addenda and Code Cases as are applicable and appropriate to meet Appendix B of 10 CFR 50 in the Federal Register, Vol. 35, No. 125. It should be noted that these material specifications are typical for the listed applications.

The welding materials used for joining the ferritic base materials of the RCPB, conform to or are equivalent to ASME Material Specifications SFA 5.1, 5.2, 5.5, 5.17, 5.18 and 5.20. They are tested and qualified to the requirements of ASME Section III rules. In addition the ferritic materials of the reactor vessel beltline are restricted to the following maximum limits of copper, phosphorous and vanadium to reduce sensitivity to irradiation embrittlement in service:

Element	Base Metal(%)	As Deposited Weld Metal(%)
Copper	0.10 (Ladle) 0.12 (Check)	0.10
Phosphorous	0.012 (Ladle) 0.017 (Check)	0.015
Vanadium	0.05 (Check)	0.05 (as residual)

The welding materials used for joining the austenitic stainless ~~stool~~ steel base materials of the RCPB conform to ASME Material Specifications SFA 5.4 and 5.9. They are tested and qualified to the requirements of ASME Section III rules.

The welding materials used for joining nickel-chromium-iron allow in similar base material combination and in dissimilar ferritic or austenitic base material combination conform to

These limits were met for all beltline materials with the exception of the Unit 1 intermediate shell forging, which exhibited a value of 0.17 percent for copper.

5.2.4.2 Acceptable Fracture Energy Levels

Initial upper shelf fracture energy levels for materials of the Reactor Vessel Beltline Region (including welds), as determined by Charpy-V-Notch Test on specimens oriented in the transverse direction of the base material, will be established for the reactor vessel irradiation surveillance test program. The surveillance program will monitor the material properties of the beltline region to assure that adequate fracture toughness is maintained.

5.2.4.3 Operating Limitations During Startup and Shutdown

Startup and shutdown operating limitations will be based on the properties of the core region materials of the reactor pressure vessel[6]. Actual material property test data will be used. The methods outlined in Appendix G to Section III of the ASME Code will be employed for the shell regions in the analysis of protection against non-ductile failure. The initial operating curves are calculated assuming a period of reactor operation such that the beltline material will be limiting. The heatup and cooldown curves are given in the technical specifications. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the adjusted reference nil-ductility temperature which includes a reference nil-ductility temperature shift (ΔRT_{NDT}).

Predicted ΔRT_{NDT} values are derived using two curves: the effect of fluence and copper and nickel content on the shift of RT_{NDT} for the reactor vessel steels exposed to 550°F temperature curve, and the maximum fluence at 1/4 T (thickness) and 3/4 T locations (tips of the code reference flaw when the flaw is assumed at inside diameter and outside diameter locations, respectively) curve. These curves are presented in the technical specifications. For a selected time of operation, this shift is assigned a sufficient magnitude so that no unirradiated ferritic materials in other components of the Reactor Coolant System will be limiting in the analysis.

The operating curves including pressure-temperature limitations are calculated in accordance with 10CFR Part 50, Appendix G and ASME Code Section III, Appendix G, requirements. Changes in fracture toughness of the core region forgings, weldments and associated heat affected zones due to radiation damage will be monitored by a surveillance program which is based on ASTM E-185-73 "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels" and 10CFR Part 50, Appendix H. The Reactor Vessel Irradiation Surveillance Program is in compliance with these documents with the exception that four of the six reactor vessel irradiation surveillance capsules will receive a fluence which is 3.6 times the maximum reactor vessel fluence. The above documents require that the capsule to vessel maximum fluence not exceed a lead factor of 3.0. At the time of the design of the surveillance program, all capsules were positioned as near to the vessel wall as possible and were limited to a fluence less than 3 times the vessel fluence. Recently a more accurate method of calculating vessel and capsule fluence has been developed which results in a lead factor of 3.6 for four of the capsules which are in violation of the above documents. This violation is not considered to be of any significant consequence since the test results from the encapsulated specimens will represent the actual behavior of the material in the vessel and therefore the evaluation of the effects of radiation on the actual vessel material will not be influenced by the larger lead factor.

The evaluation of the radiation damage in this surveillance program is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and 1/2 T compact tension specimens. The post-irradiation testing will be carried out during the life-time of the reactor vessel. Specimens are irradiated in capsules located near the core midheight and removable from the vessel at specified intervals.

The results of the radiation surveillance program will be used and nickel to verify that the ΔT_{NDT} predicted from the effects of the fluence and copper and nickel content curve is appropriate and to make any changes necessary to correct the fluence and copper and nickel curves if ΔT_{NDT} Temperature limits for preservice hydrotests and inservice leak and hydrotests will be calculated in accordance with 10 CFR Part 50, Appendix G, and ASME III, Appendix G.

5.2.4.4 The Final Rule - 10CFR50.61 specifies screening criteria of 270°F for base materials and axial welds and 300°F for circumferential welds, and requires that projected reference temperatures (RT_{PTS}s) be calculated and evaluated to these criteria. The RT_{PTS}s are projected for the inner vessel surfaces of the limiting beltline materials from the time of FSAR submittal to the expiration date of the operating license. The basis for these projections is as follows.

RT_{PTS} projections for beltline materials are sensitive to copper and nickel content and initial RT_{NDT}. ~~and less sensitive to nickel content.~~ The copper and nickel contents and initial RT_{NDT}s for units 1 and 2 beltline materials are provided in tables 5.2.11(a) and (b) in the Final Safety Analysis Report. The beltline materials consist of the intermediate forgings, lower forgings, and intermediate-to-lower forging girth welds. Based on copper and nickel content and initial RT_{NDT}, the intermediate forgings are determined to be limiting with respect to RT_{PTS} for ~~both~~ unit~~s~~ 1 ~~and 2. Therefore, RT_{PTS} projections for the other beltine materials are not necessary.~~ and to a fluence of 3×10^{19} n/cm^2 (E > 1.0 MeV) for unit 2. Above this fluence value, the circumferential weld seam between the lower and intermediate forgings is controlling for unit 2.

Insert ① → ~~The copper content, nickel content, and initial RT_{NDT} for the unit 1 intermediate forging, as reported in WCAP 9298, are 0.17 percent copper, 0.80 percent nickel, and 47°F, respectively. The copper content, nickel content, and initial RT_{NDT} for the unit 2 intermediate forging, as reported in WCAP 9455, are 0.05 percent copper, 0.78 percent nickel, and 14°F, respectively. The intermediate forging copper and~~

INSERT ①

Unit	Material Description	Cu* (%)	Ni* (%)	I-RT_{NDT}** (°F)
1	Intermediate Shell Forging 05	0.17	0.80	47
	Lower Shell Forging 04	0.08	0.83	5
	Circumferential Weld, W05	0.05	0.70	-43
2	Intermediate Shell Forging 05	0.05	0.78	14
	Lower Shell Forging 04	0.05	0.81	5
	Circumferential Weld, W05	0.05	0.70	-50

* Analysis conducted by Rotterdam Dockyard Company.

** Values based on results of Charpy tests (WCAPs 9298 and 9455) and drop weight tests performed at the Rotterdam Dockyard Company. I-RT_{NDT} is the initial RT_{NDT} prior to irradiation of the reactor vessel.

(Note: The above data is from Reference [15].)

~~nickel contents for both units are based on analyses conducted by Rotterdam Dockyard Company. The RT_{NDT}s are measured values based on results of Charpy tests performed for units 1 and 2 surveillance programs and drop weight test performed at the Rotterdam Dockyard Company~~.

The projected RT_{PTS}s are also a function of accumulated neutron fluence at the inner surface of the beltline materials. The best estimate peak neutron fluence at the inner surface for the end-of-life (expiration of operating license) is taken to be ~~2.19~~ 3.18 x 10^{19} n/cm^2 (E > 1.0 MeV) (Reference [15]). Based on a 40-year design life and an 80% capacity factor, end-of-life is taken to be 32 effective full power years (EFPY), and the best estimate peak neutron fluence at the inner surface per EFPY is ~~6.84~~ 9.94 x 10^{17}n/cm^2.

The projected RT_{PTS}s (Reference [15]) for WBN Units 1 and 2 intermediate forgings were determined by using equation 1 in 10CFR50.61, paragraph (b)(2), and the resulting values are presented below.

	RT_{PTS}(°F) 0 EFPY	3 EFPY	32 EFPY
Unit 1 - Intermediate Forging	~~95~~ 81	~~172~~ 169	~~239~~ 253
Unit 2 - Intermediate Forging	~~62~~ 48	~~80~~ 69	~~95~~ 88
Unit 2 - Circumferential Weld Seam	1	46	90

The projected end-of-life RT_{PTS}s meets the screening criteria of 270°F for forgings and 300°F for welds. Therefore, no further action is required until changes in core loadings, surveillance measurements, or other information indicate a need for updated projections.

5.2.5 Austenitic Stainless Steel

The unstabilized austenitic stainless steel material specifications used for the 1) Reactor Coolant Pressure Boundary (RCPB), 2) systems required for reactor shutdown, and 3) systems required for emergency core cooling are listed in Tables 5.2-8 and 5.2-9.

The unstabilized austenitic stainless steel material for the reactor vessel internals which are required for emergency core cooling for any mode of normal operation or under postulated accident conditions and for core structural load bearing members are listed in Table 5.2-12.

6. Hazelton, W. S., et al., "Basis for Heatup and Cooldown Limit Curves," WCAP-7924-A, April 1975.

7. Golik, M. A., "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7477-L (Proprietary), March 1970 and WCAP-7735 (Non-Proprietary), August 1971.

8. Enrietto, J. F., "Control of Delta Ferrite in Austenitic Stainless Steel Weldments," WCAP-8324-A, June 1974.

9. Shabbits, W. O., "Dynamic Fracture Toughness Properties of Heavy Section A533 Grade B Class 1 Steel Plate," WCAP-7623, December 1970.

10. "Bench Marks Problem Solutions Employed for Verification of WECAN Computer Program," WCAP-8929, June 1977.

11. K. Takeuchi, et. al., "MULTIFLEX - a Fortran-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," WCAP-8708, February 1976.

12. Nuclear Technology, Vol. 37, January 1978, "Probablistic Analysis of the Interfacing System Loss-of-Coolant Accident and Implications on Design Decisions."

13. PTS Rule, Federal Register Vol. 50, No. N1, July 23, 1985, 10 CFR 50.34.

14. NRC Policy Issue, "Pressurized Thermal Shock," SECY-82-465, November 23, 1982.

15. Chicots, J.M., "Evaluation of Pressurized Thermal Shock for Watts Bar Unit 1," WCAP-13300, April 1992, and "...Unit 2," WCAP-13301, April 1992.

TABLE 5.2-11a

WATTS BAR UNIT 1
REACTOR VESSEL TOUGHNESS DATA
(Sheet 1)

Component	Mat'l Type	Ni (%)	Cu (%)	P (%)	NDTT (°F)	Minimum 50 Ft Lb/35 Mil Temp (°F) Parallel to Major Working Direc	Minimum 50 Ft Lb/35 Mil Temp (°F) Normal to Major Working Direc	RT_{NDT} (°F)	Average Upper Shelf (Ft-Lb) Parallel to Major Working Direc	Average Upper Shelf (Ft-Lb) Normal to Major Working Direc
Closure Dome	A533-B-1		-	-	- 4	5	25*	- 4	99	64*
Closure HD Ring	A508-2		-	-	- 4	23	43*	- 4	147	96*
Head Flange	A508-2		-	-	-49	- 3	17*	-43	130	85*
Vessel Flange	A508-2		-	-	-40	-86	-66*	-40	155	101*
Inlet Nozzle	A508-2		-	-	- 4	46	66*	6	109	71*
Inlet Nozzle	A508-2				- 4	32	52*	- 4	124	81*
Inlet Nozzle	A508-2		-	-	5	45	65*	5	122	79*
Inlet Nozzle	A508-2		-	-	-13	3	23*	-13	128	83*
Outlet Nozzle	A508-2		-	-	-22	- 8	12*	-22	119	77*
Outlet Nozzle	A508-2		-	-	-31	13	33*	-27	120	78*
Outlet Nozzle	A508-2		-	-	- 4	-19	1*	- 4	141	92*
Outlet Nozzle	A508-2		-	-	-22	25	45*	-15	101	66*
Nozzle Shell	A508-2	0.87	0.12	0.005	-22	14	34*	-22	151	98*
Inter Shell	A508-2	0.80	0.17	0.012	-22	~~37~~ -5	107**	+47	~~122~~ 132	62**
Lower Shell	A508-2	0.83	0.08	0.006	+ 5	~~8~~ -60	~~60**~~ -26**	+ 5	135	~~108**~~ 88*
Shell Ring	A508-2	0.86	0.06	0.009	-40	10	30*	-30	161	105*
Bottom HD Trans Ring	A533-B-1		-	-	-31	- 4	16*	-31	128	83*
Bottom HD Trans Ring	A533-B-1		-	-	-31	-17	3*	-31	135	88*
Bottom HD Trans Ring	A533-B-1		-	-	-40	-18	2*	-40	109	71*

TABLE 5.2-11a

WATTS BAR UNIT 1
REACTOR VESSEL TOUGHNESS DATA
(Sheet 2)

Component	Mat'l Type	Ni (%)	Cu (%)	P (%)	NDTT (°F)	Minimum 50 Ft Lb/35 Mil Temp (°F) Parallel to Major Working Direc	Minimum 50 Ft Lb/35 Mil Temp (°F) Normal to Major Working Direc	RT_{NDT} (°F)	Average Upper Shelf (Ft-Lb) Parallel to Major Working Direc	Average Upper Shelf (Ft-Lb) Normal to Major Working Direc
Bottom HD Dome	A533-B-1		-	-	-31	34	54*	- 6	128	83*
Weld (INTER./LOWER SHELL)		0.70	0.05	0.010	-67	-	~~-20~~ 17	~~-61~~ -43**	-	~~122~~ 131**
Heat Affected Zone			-	-	~~-67~~ -22	-	~~-76~~ 34	~~-67~~ -22	-	~~127~~ 89**

*Estimated by the methods contained in the 'Regulatory Standard Review Plan, Section 5.3.2 Pressure Temperature Limits'
**Based on transverse data from surveillance program

TABLE 5.2-11b

WATTS BAR UNIT NO. 2 REACTOR VESSEL FRACTURE TOUGHNESS TEST DATA

COMPONENT	HEAT NO.	MAT'L. SPEC. NO.	Ni (%)	Cu %	P %	T_{NDT} °F	MINIMUM 50 FT-LB 35 MIL LAT. EXP. TEMPERATURE MWD °	NMWD °F	RT_{NDT} °F	AVERAGE SHELF ENERGY MWD FT-LB	NMWD FT-LB
Closure Head Dome	55994-1	A533B, Cl. 1		.08	~~.010~~ .006	-4	-8	12*	-4	143	93*
Closure Head Ring	07793	A508 Cl. 2		.08	~~.006~~ .008	-4	28	48*	-4	138	90*
Closure Head Flange	528994	A508 Cl. 2		.07	.009	-40	-13	7*	-40	146	95*
Vessel Flange	527944	A508 Cl. 2		.06	.009	-22	10	30*	-22	219	142*
Inlet Nozzle	528209	A508 Cl. 2		.05	.008	-22	5	25*	-22	120	78*
Inlet Nozzle	528267	A508 Cl. 2		.06	.011	-22	26	46*	-14	101	66*
Inlet Nozzle	528267	A508 Cl. 2		.06	.010	-13	23	43*	-13	93	60*
Inlet Nozzle	528329	A508 Cl. 2		.04	.009	-13	25	45*	-13	129	84*
Outlet Nozzle	528095	A508 Cl. 2		.06	.009	-22	17	37*	-22	138	90*
Outlet Nozzle	528207	A508 Cl. 2		.06	.011	-13	7	27*	-13	109	71*
Outlet Nozzle	528209	A508 Cl. 2		.05	.009	-40	5	25*	-35	128	83*
Outlet Nozzle	528329	A508 Cl. 2		.04	.009	-31	26	46*	-14	128	83*
Nozzle Shell	411572	A508 Cl. 2	0.91	.07	.005	-22	3	23*	-22	142	92*
Inter. Shell	527828	A508 Cl. 2	0.78	.05	.012	+14	~~29~~ 0	~~42**~~ 40**	14	~~132~~ 175	110**
Lower Shell	528658	A508 Cl. 2	0.81	.05	.006	5	14	~~20**~~ 38**	5	154	~~121**~~ 105**
Bottom Head Ring	528170	A508 Cl. 2	0.87	.06	.009	-40	10	30*	-30	160	104 *
Bottom Head Segment	55473-2	A533B, Cl.1		.12	.006	-31	14	34*	-26	111	72*
Bottom Head Segment	55888-2	A533B, Cl.1		.04	.010	-13	21	41*	-13	152	99*
Bottom Head Segment	55888-2	A533B, Cl.1		.04	.010	5	14	34*	5	149	97*
Bottom Head Dome	55979-2	A533B, Cl.1		.04	.011	-13	14	34*	-13	120	78*
Inter. To Lower Shell Girth Weld			0.70	.05	.010	-76	-	~~5~~ 10	~~-55~~ -50	-	~~126~~ 139**
Heat Affected Zone				-	-	-67	-	~~-47~~ -50	-67	-	130 **

* Estimated per NRC standard review plant MTB 5.3.2, "Pressure Temperature Limits."
** Based on transverse surveillance program data.

Added by Amendment 43