TVA

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

10 CFR 50.55a(a)(3)

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

Gentlemen:

In the Matter of) Docket No.50-390
Tennessee Valley Authority)

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - REQUEST FOR ADDITIONAL INFORMATION ON THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) SECTION XI ALTERNATE INSERVICE INSPECTION PROGRAM RISK-INFORMED INSERVICE INSPECTION (RI-ISI) PROGRAM (TAC NO. MB2082)

The purpose of this letter is to provide responses to NRC's request for additional information (RAI) transmitted by e-mail from L. Mark Padovan dated August 13, 2001. The additional information requested supplements the WBN submittal dated May 21, 2001, and is needed to provide consistency with Revision 1 of the submittal template issued July 2001. The submittal template was developed by the Westinghouse Owners Group (WOG) for plants that follow the WOG Methodology (WCAP-14572 Revision 1-NP-A). The revisions to the template are based on resolution of items discussed between the WOG and NRC to facilitate NRC's review of plant specific applications.

There are no regulatory commitments identified in this letter. If you have any questions concerning this additional information, please contact me at (423) 365-1824.

Sincerely,

P. L. Pace
Manager, Site Licensing
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Enclosure
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ENCLOSURE

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
REQUEST FOR ADDITIONAL INFORMATION
RISK-INFORMED INSERVICE INSPECTION (RI-ISI) RELIEF REQUEST

QUESTION 1

One major step in the WCAP process is the identification of degradation mechanisms and the development of corresponding pipe failure frequencies. The requested Table 1 summarizes the qualitative results of this step by identifying the different degradation mechanisms, combinations of mechanisms, and the prevalence of the different mechanism. The calculated ranges in Table 1 summarize the quantitative results of the analysis. This information will illustrate how the degradation mechanism identification and failure frequency development step in the WCAP methodology was implemented, and provide an overview of the results generated. Please expand the current Table 3.4-1 to include the following information.

a) System	b) Degradation Mechanism/ Combination	c) Failure Probability range at 40 years with no ISI		d)Comments
		leak	disabling leak	

a) System: Each system included in the analysis.

b) Degradation Mechanism/Combination: Segment failure probabilities are characterized in the WCAP method by imposing all degradation mechanism in a segment (even if they occur at different welds) and the worst case operating conditions at the segment on a "representative" weld, and using the resulting failure probability for the segment. Please identify the dominant degradation mechanisms and combination of degradation mechanisms selected in each system. The reported mechanisms should cover all segments in the system. The table in the current submittal is not clear about which specific degradation mechanisms or combination of mechanisms are included in the leak estimates provided.

c) Failure Probability range at 40 years with no ISI: For each dominant degradation mechanism and combination of degradation mechanisms, please provide the range of estimates developed for the leak and disabling leak sizes as applicable. If multiple LOCA sizes are estimated for larger pipes reflecting the possibility of different size leaks, the frequency range of for size should be given. The table in the current submittal provided the range of leak estimates only.

d) Comments: These should provide further explanations and clarifications on the particular characteristics of the system/segments leading to the selection of the degradation mechanism(s). Other information that should be included is the identification of which degradation mechanism(s) are applied to socket welds, if a break calculation was needed to evaluate pipe whip constraints, and if the degradation mechanism is addressed by an augmented program.

RESPONSE

The expanded Table 3.4-1 below, provides a summary of the information requested in the format agreed to with the WOG in the revised RI-ISI template.

REVISED TABLE 3.4-1
Sheet 1 of 8
FAILURE PROBABILITY ESTIMATES (WITHOUT ISI) FOR WATTS BAR UNIT 1

System	Dominant Potential Degradation Mechanism(s)/ Combination(s)	Failure Probability range at 40 years with no ISI		Comments
		Small leak*	Disabling leak (by disabling leak rate)**	
Auxiliary Feedwater [AF] System 03B	• Thermal Fatigue and Water Hammer	2.1E-05	1.6E-05	• Vibrational loadings were considered on the piping connected to the steam generators due to potential vibration caused by the reactor coolant pumps. • There is potential for thermal striping from thermal mixing of cold auxiliary feedwater (AFW) and hot feedwater (FW) at the 8-inch thermal tees in the FW by-pass piping. • In AFW (piping upstream of check valve isolating FW from AFW) check valve leakage could cause thermal striping or stratification. • Water hammer in the FW line could occur from a plant trip. AFW piping connected to the FW piping could be affected by the water hammer loading. • There is a potential for water hammer from possible gas (steam) pockets forming at AFW pumps. The AFW pump discharge piping to the steam generators could be affected by the water hammer loading. • Steam hammer could occur in the main steam piping. The piping for the turbine driven AFW pumps connected to the main steam line could be affected by the steam hammer loading.
	• Thermal Fatigue and Steam Hammer	1.0E-04	6.1E-04	
	• Thermal Fatigue, Water Hammer, and Stratification/Striping	6.8E-04 - 1.7E-03	1.8E-04 - 3.9E-04	
	• Thermal Fatigue, Water Hammer, and Erosion/Corrosion	1.0E-01	1.0E-01	
	• Thermal Fatigue, Vibratory Fatigue, Water Hammer, and Erosion/Corrosion	1.0E-01	1.0E-01	
	• Thermal Fatigue, Water Hammer, Stratification/ Striping, and Erosion/Corrosion	1.0E-01	1.0E-01	
	• Thermal Fatigue, Vibratory Fatigue, Water Hammer, Stratification/ Striping, and Erosion/Corrosion	1.0E-01	1.0E-01	

REVISED TABLE 3.4-1
Sheet 2 of 8
FAILURE PROBABILITY ESTIMATES (WITHOUT ISI) FOR WATTS BAR UNIT 1

System	Dominant Potential Degradation Mechanism(s)/ Combination(s)	Failure Probability range at 40 years with no ISI		Comments
		Small leak*	Disabling leak (by disabling leak rate)**	
Steam generator blowdown [BD] System 15	• Thermal Fatigue and Vibratory Fatigue	2.6E-04	2.0E-04	• An augmented program for flow accelerated corrosion (FAC) exists for BD piping. • Steam hammer loadings are considered on the BD piping to account for the loading from a steam generator blowdown. • Vibrational loading is considered on the piping connected to the steam generator due to potential vibration from the reactor coolant pumps.
	• Thermal Fatigue and Steam Hammer	1.0E-04 - 2.1E-03	6.1E-04 - 4.6E-03	
	• Thermal Fatigue, Steam Hammer and FAC	5.6E-01	5.6E-01	
	• Thermal Fatigue, Steam Hammer, Vibratory Fatigue, and FAC	6.2E-01	6.2E-01	

REVISED TABLE 3.4-1
Sheet 3 of 8
FAILURE PROBABILITY ESTIMATES (WITHOUT ISI) FOR WATTS BAR UNIT 1

System	Dominant Potential Degradation Mechanism(s)/ Combination(s)	Failure Probability range at 40 years with no ISI		Comments
		Small leak*	Disabling leak (by disabling leak rate)**	
Chemical & Volume Control [CH] System 62	• Thermal Fatigue	3.5E-06 - 6.7E-04	4.9E-06 - 2.8E-04	• There is a potential thermal striping and stratification from check valve leakage in the piping connected to the reactor coolant system.
	• Thermal Fatigue and Vibratory Fatigue	8.0E-06 - 8.7E-03	• SLOCA 5.5E-03 - 6.2E-03 • IE/SYS 3.8E-06 - 6.2E-03	• Flashing and cavitation at the letdown orifices could produce vibratory loadings on the adjacent piping.
	• Thermal Fatigue, Vibratory Fatigue, and Water Hammer	8.0E-06 - 8.7E-03	1.3E-05 - 5.6E-03	• Vibratory loadings were considered on the piping connected to the centrifugal charging pumps and the piping connected to the reactor coolant loop (RCL).
	• Thermal Fatigue, Vibratory Fatigue, and Striping/Stratification	7.0E-05 - 8.0E-05	• MLOCA 1.9E-05 - 6.1E-05 • SLOCA 2.4E-05 - 5.6E-05 • IE/SYS 2.8E-05 - 5.3E-05	• Potential for transient loads exists at the discharge of the high head pumps and at the discharge relief valves.
Containment Isolation [CI] System 88	• Thermal Fatigue	2.4E-05 - 8.4E-04	8.8E-07 - 3.5E-04	• Material wastage was considered for the carbon steel essential raw cooling water (ERCW) and Fire Protection piping.
	• Thermal Fatigue and Erosion / Corrosion	6.2E-02	6.2E-02	

REVISED TABLE 3.4-1
Sheet 4 of 8
FAILURE PROBABILITY ESTIMATES (WITHOUT ISI) FOR WATTS BAR UNIT 1

System	Dominant Potential Degradation Mechanism(s)/ Combination(s)	Failure Probability range at 40 years with no ISI		Comments
		Small leak*	Disabling leak (by disabling leak rate)**	
Containment Spray [CS] System 72	• Thermal Fatigue	1.1E-05 - 2.0E-04	5.6E-06 - 1.1E-04	• The potential for vibration was considered for the piping adjacent to the CS pumps and the piping adjacent to the throttling orifice. • The potential for water hammer due to gas pockets forming in the pump discharge piping was considered. The potential for water hammer in the spray header piping was considered.
	• Thermal Fatigue and Water Hammer	7.2E-06 - 2.0E-04	3.5E-06 - 1.9E-04	
	• Thermal Fatigue and Vibratory Fatigue	2.1E-05 - 2.0E-04	7.1E-06 - 5.2E-05	
	• Thermal Fatigue, Vibratory Fatigue and Water Hammer	1.5E-05 - 2.1E-04	3.0E-05 - 2.5E-04	
Feedwater System [FW] System 03A	• Thermal Fatigue, Vibratory Fatigue and Water Hammer	4.0E-04	4.8E-03	• Vibrational loadings were considered on the piping connected to the steam generators due to potential vibration caused by the reactor coolant pumps. • FAC was considered for the main FW flow path piping and the FW by-pass piping to the steam generators. The locations where high flow velocities may cause pipe wall thinning are in the FAC program. • The potential for transient loads during a plant trip was considered.
	• Erosion/Corrosion, Thermal Fatigue and Water Hammer	4.4E-02 - 1.0E-01	4.4E-02 - 1.0E-01	
	• Thermal Fatigue and Water Hammer	2.2E-04 - 4.2E-04	1.0E-03 - 5.2E-03	
	• Thermal Fatigue and Vibratory Fatigue	1.9E-04	2.1E-04	
	• Erosion/Corrosion, Thermal Fatigue, Vibratory Fatigue, Water Hammer	4.4E-02	4.4E-02	

REVISED TABLE 3.4-1
Sheet 5 of 8
FAILURE PROBABILITY ESTIMATES (WITHOUT ISI) FOR WATTS BAR UNIT 1

System	Dominant Potential Degradation Mechanism(s)/ Combination(s)	Failure Probability range at 40 years with no ISI		Comments
		Small leak*	Disabling leak (by disabling leak rate)**	
Main Steam System [MS] System 01	• Thermal Fatigue and Steam Hammer	2.7E-07 - 2.1E-04	1.3E-04 - 9.4E-04	• Vibrational loadings were considered on the piping connected to the steam generators due to potential vibration caused by the reactor coolant pumps. • The potential for transient loads during a plant trip was considered.
	• Thermal Fatigue, Steam Hammer and Vibratory Fatigue	2.7E-07	1.3E-04	
Reactor Coolant System [RC] System 68	• Thermal Fatigue	1.4E-09 - 1.8E-04	• MLOCA 2.3E-09 - 3.3E-05 • SLOCA 5.0E-11 - 1.9E-04	• Vibration was considered for the RCL piping and the piping connected to the RCL. • Thermal striping and stratification occurs in the pressurizer surge line. • Transient loads were considered to occur in the tailpipes due to steam release from pressurizer relief valves.
	• Thermal Fatigue and Vibratory Fatigue	5.8E-06 - 1.2E-02	• LLOCA 5.9E-06 - 6.9E-06 • MLOCA 5.9E-06 - 6.5E-05 • SLOCA 6.3E-06 - 7.1E-03	
	• Thermal Fatigue and Striping/Stratification	8.0E-04	• LLOCA 7.8E-04 • MLOCA 7.8E-04 • SLOCA 7.9E-04	
	• Thermal Fatigue and Water Hammer	3.7E-06 - 7.0E-05	• MLOCA 8.7E-05 - 1.3E-04 • SLOCA 9.3E-05 - 1.7E-03	
	• Thermal Fatigue, Vibratory Fatigue, and stress corrosion cracking (SCC).	3.04E-02	• MLOCA 2.6E-02 • SLOCA 2.6E-02	

REVISED TABLE 3.4-1
Sheet 6 of 8
FAILURE PROBABILITY ESTIMATES (WITHOUT ISI) FOR WATTS BAR UNIT 1

System	Dominant Potential Degradation Mechanism(s)/ Combination(s)	Failure Probability range at 40 years with no ISI		Comments
		Small leak*	Disabling leak (by disabling leak rate)**	
Residual Heat Removal System [RH] System 74	• Thermal Fatigue	9.6E-05 - 1.4E-04	1.6E-04	• System experiences temperature changes from ambient to 350°F when used for shutdown cooling. • Vibration was considered for the piping adjacent to the residual heat removal (RHR) pumps and for the piping connected to the RCL. • Transient loads were considered in the pump discharge piping due to the potential of gas intrusion in the piping. Venting is performed to remove possible gas intrusion thus reducing the probability of water hammer. • The potential for transient loads was considered on the discharge piping of the relief valves. • The potential for thermal stratification and striping from leaking check valves was considered in the 6-inch safety injection (SI) piping connected to the 14-inch RHR return line. • SCC potential was considered on one weld. The weld is located on the 6-inch SI piping connected to the 14-inch RHR return line.
	• Thermal Fatigue and Vibratory Fatigue	1.1E-07 - 1.9E-04	• LLOCA 5.3E-07 • MLOCA 5.3E-07 • SLOCA 5.3E-07 - 2.3E-04 • SYS 5.1E-07 - 7.1E-05	
	• Thermal Fatigue, Vibratory Fatigue and Water Hammer	3.3E-05 - 1.9E-04	3.0E-05 - 1.5E-04	
	• Thermal Fatigue and Water Hammer	2.6E-05 - 6.7E-04	1.3E-05 - 2.2E-04	
	• Thermal Fatigue, Vibratory Fatigue, Striping/Stratification, Water Hammer and SCC.	1.5E-03	• MLOCA 3.2E-04 • SLOCA 4.0E-04 • SYS 4.0E-04	

REVISED TABLE 3.4-1
Sheet 7 of 8
FAILURE PROBABILITY ESTIMATES (WITHOUT ISI) FOR WATTS BAR UNIT 1

System	Dominant Potential Degradation Mechanism(s)/ Combination(s)	Failure Probability range at 40 years with no ISI: Small leak*	Failure Probability range at 40 years with no ISI: Disabling leak (by disabling leak rate)**	Comments
Safety Injection System [SI] System 63	• Thermal Fatigue	1.7E-05 - 3.8E-04	6.1E-07 - 1.3E-04	• Vibration was considered for the piping adjacent to the SI pumps and for the piping connected to the RCL. • NRC Bulletin 88-08 identified potential thermal stratification/striping concerns for piping connected to the RCL. • Transient loads were considered in the SI pump discharge piping due to potential gas pockets. The potential for gas pockets (due to nitrogen coming out of solution) at high points in the piping exists due to back-leakage through check valves. • The potential for transient loads was considered on the discharge piping of the relief valves. • SCC potential was considered on 7 welds. Two welds are located on 8-inch RHR pump discharge piping inside containment. Four welds are located on 16-inch and 24-inch piping from the RWST. One weld is located on 8-inch piping which interconnects the SI pump suction piping with the RHR heat exchanger discharge piping.
	• Thermal Fatigue and Water Hammer	5.4E-07 - 2.3E-04	9.6E-06 - 2.0E-04	
	• Thermal Fatigue, Vibratory Fatigue	5.6E-05 - 1.0E-04	1.6E-05 - 7.9E-05	
	• Thermal Fatigue, Water Hammer and Striping / Stratification	2.1E-04 - 3.7E-04	1.2E-04 - 1.6E-03	
	• Thermal Fatigue, Vibratory Fatigue and Water Hammer	1.9E-05 - 9.0E-04	2.1E-05 - 3.3E-04	
	• Thermal Fatigue, Vibratory Fatigue, Water Hammer and Striping/ Stratification	1.3E-06 - 1.0E-02	• LLOCA 2.8E-05 • MLOCA 1.5E-05 - 2.7E-05 • SLOCA 1.5E-05 - 6.4E-03 • SYS 9.9E-06 - 8.7E-03	
	• Thermal Fatigue and SCC	1.6E-01	4.2E-02	
	• Thermal Fatigue, Water Hammer, and SCC	8.5E-04	3.8E-05	
	• Thermal Fatigue, Vibratory Fatigue and SCC	8.8E-02	3.9E-02	

REVISED TABLE 3.4-1
Sheet 8 of 8
FAILURE PROBABILITY ESTIMATES (WITHOUT ISI) FOR WATTS BAR UNIT 1

System	Dominant Potential Degradation Mechanism(s)/ Combination(s)	Failure Probability range at 40 years with no ISI		Comments
		Small leak*	Disabling leak (by disabling leak rate)**	
Water Quality and Sampling [SQ] System 43	• Thermal Fatigue and Water Hammer	1.5E-04	2.2E-04	• Water hammer is a potential concern for the piping connected to the RH system.

Notes:
* Small leak is equal to a through-wall-crack.
** Disabling leak rate - LLOCA (Large LOCA), MLOCA (Medium LOCA), SLOCA (Small LOCA), and SYS (system disabling leak). When no leak rate is shown, this is the system disabling leak rate.

ENCLOSURE

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
REQUEST FOR ADDITIONAL INFORMATION
RISK-INFORMED INSERVICE INSPECTION (RI-ISI) RELIEF REQUEST

QUESTION 2

Another major step in the WCAP process is assignment of segments into safety significance categories based an integrated decision making process, and the selection of segments for inspection locations. The requested Table 3 summarizes the results of the safety significance categorization process as determined by the quantitative criteria, by the expert panel's deliberation on the medium safety significant segments, and by the expert panel's deliberations based on other considerations. The summarizing information requested in Table 3 will provide an overview of the distribution of the safety significance of the segments based on the quantitative results, and the final distribution based on the integrated decision making. Each segment has four RRWs calculated, a CDF with and without operator action, and a LERF with and without operator action. Please provide the following Table.

System	Number of segments with any RRW >1.005	Number of segments with any RRW between 1.005 and 1.001	Number of segments with any RRW between 1.005 and 1.001 placed in HSS	Number of segments with all RRW < 1.001	Number of segments with all RRW < 1.001 selected for inspection	Total number of segments selected for inspection

RESPONSE

Table 3 below provides the risk reduction worth (RRW) information requested. Table 3 relates to Table 3.7-1 in the revised submittal template.

TABLE 3
(TABLE 3.7-1)
SUMMARY OF RISK EVALUATION AND EXPERT PANEL CATEGORIZATION RESULTS
WATTS BAR NUCLEAR PLANT UNIT 1

System	Number of Segments with any RRW ≥1.005	Number of segments with any RRW <1.005 and ≥1.001	Number of segments with any RRW <1.005 and ≥1.001 placed in HSS	Number of Segments With all RRW <1.001	Number of segments with all RRW <1.001 selected for inspection	Total number of segments selected for inspection (High Safety Significant Segments)
AF	4	12	0	7	0	4
BD	8	8	4	17	0	12
CH	16	7	3	86	0	13
CI	0	0	0	116	0	0
CS	0	17	0	8	1	1
FW	6	36	6	29	0	12
MS	0	4	0	9	0	0
RC	7	65	15	52	0	22
RH	0	14	1	12	4	5
SI	15	48	30	52	8	53
SQ	0	4	0	2	0	0
Total	56	215	59	390	13	122

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WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
REQUEST FOR ADDITIONAL INFORMATION
RISK-INFORMED INSERVICE INSPECTION (RI-ISI) RELIEF REQUEST

QUESTION 3

Another major step in the WCAP process is development of the consequences of segment ruptures. The WCAP methodology requires that a summary of the consequences be developed for each system and provided to the expert panel during their deliberations. Please provide this summary for each system. The summary will illustrate that the appropriate types of consequences (i.e., initiating events, mitigating system failure, and combinations) are included in the evaluation and will provide an overview of the results of the step.

RESPONSE

Below is a table that summarizes the appropriate types of consequences for each system included in the evaluation. In the revised WCAP submittal template, this table corresponds to Table 3.3-1.

TABLE 3.3-1
SUMMARY OF POSTULATED CONSEQUENCES BY SYSTEM

SYSTEM	SUMMARY OF CONSEQUENCES
AF - Auxiliary Feedwater	The direct consequences postulated from piping failures from this system are feedline/steamline breaks, failure of up to two trains of AFW system, and loss of the condensate storage tank (CST). Indirect effects were postulated for five AFW segments in which the normal and alternate steam supplies to the AFW turbine-driven AFW (TDAFW) pump result in a steam line break in the south valve vault resulting in the loss of steam generator (SG) 1 and 4 power operated relief valves (PORVs).
BD - Steam Generator Blowdown	The direct consequences postulated from piping failures from this system include steam line breaks inside and outside containment, loss of normal and alternate steam supply to the TDAFW pump, and failure to isolate the system on a steam generator tube rupture.
CH - Chemical & Volume Control	The direct consequences associated with piping failures are reactor trip on low seal injection flow, small loss-of-coolant-accident (LOCA), loss of one or both centrifugal charging pump (CCP) trains for injection, recirculation, and emergency boration, loss of refueling water storage tank (RWST) refill function, loss of RWST outside containment and loss of containment sump recirculation outside containment and outside the polar crane wall inside containment.
CI - Containment Isolation (Supersystem)	The direct consequences postulated from piping failures from this system are failures of the containment isolation system, failure to isolate the containment purge system, failure of one train of auxiliary control air, loss of primary makeup water pumps, loss of reactor coolant pump (RCP) thermal barrier cooling and loss of component cooling water.

TABLE 3.3-1
SUMMARY OF POSTULATED CONSEQUENCES BY SYSTEM

SYSTEM	SUMMARY OF CONSEQUENCES
CS - Containment Spray	The direct consequences associated with piping failures are the loss of containment spray, loss of residual heat removal (RHR) train spray headers, loss of the RWST outside containment and loss of containment sump recirculation outside containment. An indirect consequence is postulated for one segment in which a piping failure could spray and thus fail the primary water makeup pumps.
FW - Feedwater	The direct consequences postulated from piping failures from this system are loss of main feedwater restoration, loss of the normal and alternate steam supplies to the TDAFW pump, feedline breaks inside and outside containment, and steam flow/feedwater flow mismatch resulting in a plant trip.
MS - Main Steam	The direct consequences postulated from piping failures from this system are loss of normal and alternate supply to the TDAFW pump, steam line break inside and outside containment, loss of steam dumps and SG PORVs. Indirect effects were postulated for two segments in which a steam line break in the south valve vault results in the loss of SG 1 and 4 PORVs and the TDAFW pump. For two other segments a main steam line break in the north valve vault is postulated to result in the loss of SGs 2 and 3 PORVs.
RC - Reactor Coolant	The direct consequences associated with piping failures are large, medium and/or small LOCAs and loss of emergency core cooling system (ECCS) flow to one loop.
RH - Residual Heat Removal	The direct consequences associated with piping failures are the loss of one or both RHR trains for normal shutdown cooling and low pressure injection and recirculation, loss of RWST outside containment and outside the polar crane wall inside containment and loss of containment sump recirculation outside containment. Three segments involve LOCA initiating events (large, medium and small LOCAs).
SI - Safety Injection	The direct consequences associated with piping failures are the loss of accumulator injection, loss of one or both SI trains for injection and recirculation from either the charging system or SI system, loss of RWST outside containment and outside the polar crane wall inside containment and loss of containment sump recirculation outside containment and outside the polar crane wall inside containment. Several segments involve LOCA initiating events (large, medium and small LOCAs). An indirect consequence is postulated for one piping segment in which a piping failure could spray and thus fail the primary water makeup pumps.
SQ - Water Quality and Sampling	The direct consequences postulated from piping failures from this system are the loss of sump water inventory outside containment and outside the polar crane wall inside containment.

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QUESTION 4

Please add the statement that the sensitivity study to address uncertainty as described on page 125 was performed, and identify how many segments' RRW increased from below 1.001 to greater than or equal to 1.005. If the sensitivity study was not performed, provide a description and justification of any deviation.

RESPONSE

The uncertainty analysis as described on WCAP page 125 and 129 was performed and included as part of the base process. The results of these calculations were provided in Table 3.5-1 of TVA's letter dated May 21, 2001. Given that the uncertainty was directly incorporated into the risk evaluation calculations, any chance that segments with RRWs below 1.001 moved to greater than or equal to 1.005 is already captured. For this request for additional information, a comparison was performed between the RRW values before and after the uncertainty analysis. The comparison results indicated that no segments had an increase in RRW value from less than 1.001 to greater than or equal to 1.005.

QUESTION 5

Please state that the change in risk calculations were performed according to all the guidelines provided on page 213 of the WCAP or provide a description and justification of any deviation.

RESPONSE

The change in risk calculations were performed according to the guidelines provided on page 213 of the WCAP with one exception. Per the WCAP, credit is taken for leak detection for the reactor coolant system (RCS) piping segment failure probabilities. The change in risk methodology used for WBN extended this to segments located inside containment and that interface with the RCS (systems such as SI, CVCS, RHR, etc.) such that radiation monitors and sump level detects a leak. For these segments, the failure probability "with ISI" for those being inspected by non-destructive examination (NDE) and without ISI for those not being inspected, is used along with credit for leak detection.

QUESTION 6

The quantitative change in risk results are adequately summarized in the current template tables 3-5 and 3-10. Please state that all four criteria for accepting the final selection of inspection locations provided on page 214 and 215 in WCAP-14572 Revision 1-NP-A were applied. If all four criteria were not used, please provide a description and justification of the deviation. If comparison with any of the criteria indicated that "reevaluation" of the selected locations was needed, please identify the criteria that required the reevaluation and summarize the results of the reevaluation. If the results of the reevaluation can be found in the footnotes of Table 5-1, please refer to the footnotes.

RESPONSE

The change in risk calculations were performed according to the guidelines provided on pages 210 through 215 of the WCAP. The approach evaluated the change in risk with the inclusion of the probability of detection as determined by the Structural Reliability and Risk Assessment (SRRA) model. Adjustments were made to add segments until the four criteria for accepting the results discussed on page 214 and 215 in the WCAP were met. This evaluation resulted in the identification of 15 piping segments for WBN which examinations are now required (systems identified in Table 5-1 of the WBN RI-ISI submittal dated May 21, 2001, via a footnote).

The three criteria requiring reevaluation of the selected locations per WCAP-14572 Revision 1-NP-A (page 214) are:

1. The total change in piping risk in moving from Section XI to RI-ISI.
2. An evaluation of the dominant system contributors to the total risk (where a system contribution is "dominant" when the contribution to the total is greater than 10 percent).
3. The risk increase for a given system in core damage frequency (CDF) and large early release frequency (LERF) in moving from the current Section XI program to the RI-ISI program.

In the change-in-risk calculations, several systems initially contributed to unacceptable risk increases. The following changes were made to meet the acceptance criteria:

- Auxiliary Feedwater had an unacceptable risk increase for CDF without operator action. Four additional segments were added to meet Criterion 3.
- Chemical and Volume Control had unacceptable risk increases for CDF without operator action and LERF without operator

action. One additional segment was added to meet Criteria 1 and 2.

- Containment Spray had unacceptable risk increases for CDF without operator action and LERF without operator action. Two segments were added to meet Criterion 3.
- Residual Heat Removal had unacceptable risk increases for CDF without operator action and LERF without operator action. Three segments were added to meet Criterion 3.
- Feedwater had an unacceptable risk increase for CDF without operator action, CDF with operator action, LERF without operator action, and LERF with operator action. Five segments were added to meet Criteria 2 and 3.

QUESTION 7

Briefly describe the qualifications, experience, and training of the users of the SRRA code on the capabilities and limitations of the code.

RESPONSE

An engineering team was established that has access to expertise from ISI, NDE, materials, stress analysis and system engineering. Each of the team members has at least 10 years of experience in their respective disciplines. The team is knowledgeable of the potential piping degradation mechanisms and loading conditions associated with the operation of WBN Unit 1.

The team was trained in the failure probability assessment methodology and the Westinghouse structural reliability and risk assessment (SRRA) code, including identification of the capabilities and limitations as described in WCAP-14572, Revision 1-NP-A, Supplement 1. The training was performed in both classroom and hands-on sessions. Technical reviews of the SRRA calculations were performed by both the TVA and Westinghouse engineers who are cognizant of the structural reliability requirements. Continuous guidance was provided by cognizant Westinghouse SRRA specialists throughout the duration of the development of the RI-ISI Program for WBN Unit 1.

ENCLOSURE

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
REQUEST FOR ADDITIONAL INFORMATION
RISK-INFORMED INSERVICE INSPECTION (RI-ISI) RELIEF REQUEST

QUESTION 8

Please provide the following information regarding the treatment of augmented programs during the RI-ISI program development.

a) Treatment of augmented program inspections during categorization is described on page 80 (Section 3.5.5) of WCAP-14572, Rev. 1-NP-A. Please add the statement that the effects of ISI of existing augmented programs are included in your calculations used to categorize the segments or provide a description and justification of any deviation.

b) When the SRRA code is used for calculating failure probabilities for FAC, please describe if calculations were coordinated with the existing plant program since the code requires input that can be obtained from the knowledge gained from ongoing monitoring and evaluations of wall thinning rates.

RESPONSE

a) The effects of ISI of existing augmented programs are included in the risk evaluation used to assist in categorizing the segments as described on pages 80 and 105 of WCAP-14572. There were no deviations. The failure probabilities used in the risk-informed process are documented and maintained in the plant records.

b) When the SRRA code is used for calculating failure probabilities, the data used for FAC was coordinated with the existing plant program. The locations of piping subject to FAC and the associated wastage values were obtained from the FAC representative of the Corporate Material Engineering group. The FAC representative coordinated with the plant personnel responsible for the ongoing monitoring programs to determine the wall thinning rates. Information provided by the FAC representative is documented in the SRRA calculations.

QUESTION 9

Please confirm that SRRA code was only used to calculate failure probabilities for the failure modes, materials, degradation mechanisms, input variables and uncertainties it was programmed to consider as discussed in the WCAP Supplement 1, page 15. For example, SRRA code should only be applied to standard piping geometry (circular piping geometry with uniform wall thickness).

If the code was applied to any non-standard geometry, please describe how the SRRA inputs were developed.

RESPONSE

The SRRA code was only used to calculate failure probabilities for the failure modes, materials, degradation mechanisms, input variables, and uncertainties it was programmed to consider as discussed in the WCAP Supplement 1. The SRRA code was only applied to standard piping geometry (circular piping geometry with uniform wall thickness). The SRRA code was not applied to any non-standard geometry. Each SRRA calculation lists the limitation of the SRRA code for the piping in the TVA RI-ISI Program and provides a statement that the limitations did not exist for the piping segments evaluated in the calculation.

QUESTION 10

Please describe any sensitivity studies performed to support the use of the SRRA code.

RESPONSE

During and after the SRRA training provided by Westinghouse, the effects of various input parameters, such as SCC potential, were investigated via unofficial SRRA runs that are not part of the RI-ISI Program database. During the RI-ISI Program most sensitivity studies were performed to determine the design limiting conditions for pipe break conditions. The following events and their probability of occurrence were considered in sensitivity studies as appropriate:

- normal operation,
- potential water hammer events,
- potential faulted design events such as a seismic event, and
- potential failure of snubbers.

The SRRA code was then used to calculate failure probabilities for the failure modes, materials, degradation mechanisms, input variables, and uncertainties, the code was programmed to consider. The failure probabilities of the SRRA evaluations were reviewed to determine if the probabilities represented expected results for the plant conditions evaluated. If necessary, the SRRA input parameters were adjusted to account for actual plant conditions such as vibration for systems which do not operate continuously, potential thermal stratification and striping concerns addressed in NRC Bulletins 88-08, "Thermal Stresses in RCS Piping," and 88-11, "Pressurizer Surge Line Thermal Stratification," material wastage or other input parameters which were judged to either over or under estimate the failure probabilities of a segment.

QUESTION 11

Please provide the total number of Class 1 butt welds and socket welds, the percentage of Class 1 butt welds selected for volumetric inspection, and the percentage of Class 1 socket welds selected for inspection in the RI-ISI Program. If the total number of socket welds is not readily available, an estimate of the number is acceptable.

RESPONSE

The WBN Unit 1 ISI Program utilizes Paragraph IWB-1220 of the 1989 Edition of ASME Section XI. This paragraph permits Class 1 piping of 1-inch nominal pipe size (NPS) and smaller to be exempt from surface and volumetric examination. Therefore, weld counts for piping 1-inch NPS and smaller are not tracked for the ISI Program.

WBN Unit 1 includes 468 Class 1 circumferential butt welds greater than 1-inch NPS and 31 Class 1 branch connection welds greater than 2-inch NPS which results in a total of 499 Class 1 butt welds (weld numbers are based on plant procedure 1-TRI-0-10 Revision 6, "ASME Section XI ISI/NDE Program"). Branch connection welds less than or equal to 2 inches NPS are covered by Request for Relief, 1-RI-ISI-2, and are not included in the above totals. Forty-four Class 1 butt welds have been selected for volumetric examination for the WBN Unit 1 RI-ISI Program. This results in 8.8% of these 499 Class 1 butt welds being selected for volumetric examination.

WBN Unit 1 includes 467 Class 1 socket welds greater than 1-inch NPS (weld numbers are based on plant procedure 1-TRI-0-10 Revision 6, "ASME Section XI ISI/NDE Program"). TVA has not identified any active or postulated piping failure mechanisms that initiate from the outside diameter. Volumetric examinations of socket welds for active or postulated piping failure mechanisms that initiate from the inside diameter are not practical and are covered by Request for Relief 1-RI-ISI-2. High safety significant segments that contain socket welds will be VT-2 examined during system pressure tests. Socket welds within the segment will be VT-2 examined. Therefore, specifying a percentage of Class 1 socket welds selected for examination for the RI-ISI program would not be meaningful.

QUESTION 12

Page 83 of the Topical states that for a WOG plant application, "(SRRA) tools were used to estimate the failure probabilities for the piping segment". Page 6 and 7 of the related safety evaluation also state that the failure probability estimate, "is subsequently used to represent the failure probability of the segment." Please confirm that, where the SRRA code was applicable, the appropriate failure frequencies estimated by the SRRA code were used in the subsequent risk ranking and change in risk calculations. If, instead, the failure frequencies used in the risk ranking or the change in risk calculations were selected from a range of values (or otherwise modified) by the expert panel or other analysts, please provide a description of this process and explain how your method comports with the approved Topical and the SE.

RESPONSE

The failure probabilities for WBN were calculated using the Westinghouse Windows version of the SRRA software program. These failure probabilities were used directly as inputs into the subsequent risk ranking and change in risk calculations. As such, no deviation from the methodology described in WCAP-14572, Revision 1-NP-A was made.