TVA

Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

October 25, 1999

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

Gentlemen:

In the Matter of	)	Docket No. 50-296
Tennessee Valley Authority	)	

BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 3 - SUPPLEMENTAL INFORMATION REGARDING THE PROPOSED RISK INFORMED INSERVICE INSPECTION (RI-ISI) PROGRAM (MA5355)

The purpose of this letter is to supplement the proposed BFN Unit 3 RI-ISI program that was submitted to NRC by TVA's letter dated April 23, 1999. The proposed program contains an alternative to the current American Society of Mechanical Engineers Section XI inservice inspection requirements for Code Class 1, 2, and 3 piping. In a meeting notice dated August 31, 1999, the staff identified a list of draft questions to be addressed in a meeting scheduled for September 20, 1999.

On September 20, 1999, TVA and Enertech personnel met with the staff regarding the BFN Unit 3 RI-ISI program submittal. During the meeting, TVA addressed each of the staff's questions listed in the August 31, 1999, meeting notice. Based on some of TVA's responses, the staff requested additional information which could not be provided at the meeting. TVA agreed to provide additional information for the following questions from the August 31, 1999, meeting notice.

- Probabilistic Safety Assessment Branch Question 14, Add a statement of changes in Core Damage Frequency and Large Early Release Frequency to RI-ISI program.
- Probabilistic Safety Assessment Branch Question 15, Address the consequences of excluding portions of systems.

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- Probabilistic Safety Assessment Branch Question 24, Revise Table 3.7-2, risk reduction worth (RRW) value for segment 3-068-009.
- Materials and Chemical Engineering Branch Question 1, Revise Table 5-1 to show the number of dual credit examinations by system and class.
- Materials and Chemical Engineering Branch Question 4, Add a statement regarding a deviation from the Westinghouse Owners Group methodology.

In addition to the above, the staff requested that TVA provide the following information not related to the questions in the August 31, 1999, meeting notice.

- Further describe the process discussed in Section 4 of the RI-ISI program submittal to discuss how feedback of new relevant information to the program will be appropriately adjusted.
- Address weaknesses in the Unit 3 PRA as described in the staff's letter to TVA dated May 4, 1999, Browns Ferry, Unit 3 Individual Plant Examination Generic Letter 88-20 (TAC NO. M74384).

The enclosure contains the staff's requested information and the attachment to the enclosure provides the revised BFN Unit 3 RI-ISI program pages. There are no commitments contained in this letter. If you have any questions, please telephone me at (256) 729-2636.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosure
cc: See page 3

Enclosure
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ENCLOSURE

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 3

TVA's SUPPLEMENTAL INFORMATION REGARDING THE RISK-INFORMED INSERVICE INSPECTION (RI-ISI) PROGRAM

I. Background

On April 23, 1999, TVA submitted the BFN Unit 3 RI-ISI program to NRC. The proposed program contains an alternative to the current American Society of Mechanical Engineers Section XI inservice inspection requirements for Code Class 1, 2, and 3 piping. In a meeting notice dated August 31, 1999, the staff identified a list of draft questions to be addressed in a meeting scheduled for September 20, 1999.

On September 20, 1999, TVA and Enertech personnel met with the staff regarding the BFN Unit 3 RI-ISI program submittal. During the meeting, TVA addressed each of the staff's questions listed in the August 31, 1999, meeting notice. Based on some of TVA's responses, the staff requested additional information which could not be provided at the meeting. The specific questions listed in the staff's August 31, 1999, meeting notice in which TVA stated that additional information would be provided are below under items II through VI. In addition, information requested by the staff which was unrelated to the questions in the meeting notice is included under items VII and VIII. The attachment to this enclosure provides the BFN Unit 3 RI-ISI revised program pages.

II. Probabilistic Safety Assessment Branch Question 14

In Table 3.10-2, no operator actions, your CDF decreased by 2.2E-6/yr and your LERF decreased by 1.8E-6/yr. Since you are only adding one inspection not previously being done, please explain where the decreases in CDF and LERF are coming from.

TVA's Meeting Response

TVA stated that Table 3.10-2 indicates the amount of CDF/LERF that is detected by the subject program, not the decrease in CDF/LERF.

Staff's Conclusion/Requested Information

The staff agreed that TVA's response is acceptable, however, requested that TVA provide the changes in CDF/LERF due to the Risk-Informed ISI program.

TVA's Supplemental Response

The change in CDF due to the Risk-Informed ISI program is a reduction of 3.8E-07 and the change in LERF due to the Risk-Informed ISI program is a reduction of 1.08E-07. Refer to the attachment for the revised BFN Unit 3 RI-ISI program, Page E-28.

III. Probabilistic Safety Assessment Branch Question 15

Page E-9 discusses "portions of a system" to be included. Which systems had portions excluded? What is the basis for excluding portions? Are any inspections currently performed in the excluded portions and, if so, what happens to these inspections?

TVA Meeting Response

TVA stated that section 3.2 of WCAP-14572 provides three criteria for selecting systems:

- *All Class 1, 2, and 3 systems currently within the ASME Section XI program;*
- *Piping systems modeled in the PSA; or*
- *Various balance of plant fluid systems determined to be of importance (mainly based on Maintenance Rule ranking).*

These criteria were applied as described in the first paragraph of 3.1 of the submittal. The basis for determination of importance with respect to the Maintenance Rule was Appendix B to Browns Ferry procedure 0-TI-346, which describes each system and defines the portion of each system to be considered significant. The portions excluded are:

- *Those portions of Condensate and Demineralized water that do not a) provide a water source and heat sink for EOPs and to mitigate accidents, or b) condense steam from the reactor vessel/main turbine, or c) deliver water to the suction of feedwater, or d) provide stored condensate and a flow path for use by HPCI and/or RCIC.*
- *Those portions of Condenser Circulating Water that do not provide cooling water to the main condenser to condense steam.*

There are no current inspections performed in the excluded portions.

Staff Conclusion/Requested Information

The staff agreed that TVA's response is acceptable; however, requested that TVA address the consequences of excluding portions of systems.

TVA's Supplemental Response

The excluded portions of the Condensate and Demineralized Water are the internals of the demineralizers, one inch oxygen injection piping, and the normally isolated drain from the drain cooler inlets, none of which affect the required functions. For Condenser Circulating Water, the excluded portions are the buried pipe discharge from the condensers. Failure of these portions of the systems would not result in any loss of function or flooding of important equipment.

IV. Probabilistic Safety Assessment Branch Question 24

Comparing Table 3.7-3 to 3.7-2, for the recirculation system, it would appear that there should be four medium and three low risk segments categorized based on RRW values rather than the three medium and four low shown in Table 3.7-2.

TVA's Meeting Response

TVA stated that there is a typographical error in Table 3.7-2. RRW for 3-068-009 should be 1.000. There is no impact on designation of the segment as HSS.

Staff's Conclusion/Requested Information

The staff agreed that TVA's response is acceptable, but requested a correction to the table.

TVA's Supplemental Response

Refer to the attachment for the revised RI-ISI program Table 3.7-2, RRW for segment 3-068-009, Page E-19

V. Materials and Chemical Engineering Branch Question 1

Section 2.2 states that augmented program inspections, with the exception of IGSCC Category "A" welds, remain unchanged. Please describe if the augmented inspection program inspections are credited toward the samples required using the RI-ISI sample selection criteria. If so, please specify the percentage of inspections in the RI-ISI program that are included in the current augmented inspection programs. Also, Section 3.8 states that all locations identified for examination are locations already identified under existing programs, either Section XI, IGSCC, or FAC. The staff has a concern that the following issues should be addressed:

- The inspection samples should include a reasonable representation of material conditions (e.g., stainless steel and carbon steel).

TVA's Meeting Response

TVA stated that the samples include 16 carbon steel segments and 22 stainless steel segments.

Materials and Chemical Engineering Branch Question 1 (continued)

- Each degradation mechanism type existing in high safety significance (HSS) locations should be inspected.

TVA Meeting Response

TVA stated that each degradation mechanism that contributes to the significance of a High Safety Significance segment is inspected in this program.

Materials and Chemical Engineering Branch Question 1 (Continued)

- Typically no more than one half of the N-577 inspections should be taken from the augmented inspection program.

TVA Meeting Response

TVA stated that only the examinations for FAC are credited toward the RI-ISI examinations. These represent 15/84 examinations (17.8%). While the examinations for IGSCC utilize a method which must be demonstrated to comply with the requirements of NUREG-0313, these examinations are conducted under all rules applying to the Section XI program, and are considered part of the Section XI Program.

Staff's Conclusion/Requested Information

The staff agreed that TVA's response is acceptable, however, requested that Table 5-1 of the submittal be revised to show the number of "dual credit" examinations by system and class.

TVA's Supplemental Response

Refer to the attachment for the revised RI-ISI program Table 5-1, that shows the number of "dual credit" examinations by system and class, Page E-30.

VI. **Materials and Chemical Engineering Branch Question 4**

The segment definition in Section 3.2 is based on a combination of consequence and failure probability. Although acceptable, this is a deviation from the WOG methodology and should be listed as such.

TVA's Meeting Response

TVA agreed with the staff and stated that this will be provided in a follow-up submittal.

TVA's Supplemental Response

The definition provided in Section 3.2 allowed use of failure probability to further define piping segments. While consideration was given to changes in failure probability, there was no impact on segment boundaries;

therefore, the option was not used and is not needed. The definition is being revised to eliminate this option. Refer to the attachment for the revised RI-ISI program, Page E-10. This eliminates the need to identify this as deviation from the WOG methodology.

VII. Staff's Additional Information Request

Further describe the process discussed in Section 4 of the RI-ISI program submittal to discuss how feedback of new relevant information to the program will be appropriately adjusted.

TVA's Response

Program Implementation

The RI-ISI program will be maintained and adjusted for new relevant information. The control process to adjust the RI-ISI program will include the following inputs:

- Changes to plant design features.
- Changes to plant procedures and PSA.
- Equipment performance changes.
- Information on individual plant and industry failures.
- Examination results

During each operating cycle, the Program Owner will maintain an awareness of input changes. The BFN site control processes that provide input into the RI-ISI program will be enhanced to include the appropriate guidance. After each refueling outage, the effects of the changes will be evaluated to determine if a change to the Program is required.

The RI-ISI program will be updated, if required, before the next refueling outage. The Maintenance Rule Expert Panel will review proposed RI-ISI program changes and provide program oversight. The following provides an overview of the RI-ISI program inputs.

Changes to Plant Design Features

Design changes have the potential to change piping configuration and alter stress calculations which were used as input to the calculations performed in support of the RI-ISI program. New systems and branch piping will be evaluated for inclusion into the scope of the RI-ISI program. Consequently, the Design Control program will be revised to recognize RI-ISI and to ensure impact is appropriately evaluated during design preparation, review, and implementation. The existing design impact review process will also be used to ensure the impact of design changes on RI-ISI has been appropriately considered prior to final approval. The calculations supporting the RI-ISI program will be entered into TVA's calculation tracking program to ensure appropriate predecessors and inputs are identified and considered during design change preparation and review.

Changes to PSA

Since the PSA forms the basis for the RI-ISI program, any changes to the PSA or risk significance determination will be evaluated for impact on the RI-ISI program. This would also include changes to risk significance categories mandated by the Maintenance Rule Expert Panel. The PSA and Design Control procedures will be revised to ensure PSA changes also consider changes to the RI-ISI program and that RI-ISI changes are initiated as required.

Changes to Plant Procedures

Changes to plant procedures that affect ISI, such as system operating parameters, test interval, or the ability of plant operations to perform actions associated with accident mitigation will be evaluated for effect on the program.

Equipment Performance Changes

Equipment performance changes will be reviewed with appropriate plant personnel (e.g., system engineers, maintenance etc.,) to ensure that changes in performance parameters (e.g., valve leakage, increased pump testing, vibration problems) are considered in the RI-ISI update. Adverse equipment performance will be evaluated for changes to the RI-ISI inspection scope.

Information on Individual Plant and Industry failures

The Program Owner will consider applicable piping failures or degradations identified by the site's corrective action program. Industry awareness will be maintained through the sites Operating Experience program, NRC Generic Letters and Bulletins, site participation in Boiling Water Owners Group initiatives, and participation in the ASME Section XI Code committee activities.

Examination Results

NDE examinations, pressure tests, and corresponding VT-2 visual examinations for leakage that are determined to have unacceptable flaws, evidence of service related degradation or indications of leakage will be evaluated for effect to the program.

RI-ISI Program Review

The Maintenance Rule Expert Panel will provide the oversight role for the RI-ISI program. The Expert Panel will review proposed changes to the program.

As with past reviews, personnel possessing expertise in RI-ISI evaluation and ISI inspection/evaluation will be present during presentation and review of the above items.

VIII. Staff's Additional Information Request

The staff requested TVA to address weaknesses in the Unit 3 PSA as described in the staff's letter to TVA dated May 4, 1999, Browns Ferry, Unit 3 Individual Plant Examination (IPE) Generic Letter 88-20 (TAC NO. M74384)

TVA's Response

The May 4, 1999, staff Evaluation (SE) approved the BFN Unit 3 PSA for the purpose of documenting compliance with GL 88-20. The SE identified weaknesses in three areas as follows:

- Lack of discussion of insights gained from the IPE analysis as requested in NUREG 1335, Individual Plant Examination Submittal Guidance.
- Lack of input for the "front end" topics requested in NUREG 1335.
- Lack of "back end" evaluation of containment performance including large early release frequency.

It should be noted that all but one of the above weaknesses are related to documentation that was requested by the staff in NUREG 1335 for PSA submittals in response to GL 88-20. This documentation was not available to the staff because TVA never submitted the

Unit 3 PSA in response to GL 88-20 nor did TVA intend for the Unit 3 PSA to be used for that purpose.

The staff originally approved the BFN Unit 2 PSA for the purpose of meeting GL 88-20 in a SE dated September 28, 1994. In that SE, the staff stated that GL 88-20 would remain open pending receipt and review of a "Multi Unit PRA" (MUPRA) that would specifically address the shared systems between the three BFN Units. Through several meetings and letters, the staff and TVA agreed the MUPRA would be of limited scope and would only address ten shared systems and two initiating events. TVA submitted the MUPRA to NRC on April 14, 1995.

TVA, on its own initiative, subsequently developed the Unit 3 PSA for a number of reasons (including the Maintenance Rule). The Unit 3 PSA was identical to the Unit 2 PSA except for differences in design and operating data between the Units. Also, due to the intended use of the Unit 3 PSA it did not include a Level 2 (containment response) analysis.

As a direct consequence of using the Unit 3 PSA model for various risk-informed applications, TVA updated the Unit 3 model to include the Level 2 analysis. The LERF values were provided to the staff in the Risk-Informed ISI program submittal.

As explained above, the weaknesses identified by the staff in the May 4, 1999, SER applied primarily to documentation. The reason for this as explained above, was that the Unit 3 PSA was never submitted to the staff by TVA for the purpose of compliance with GL 88-20. This lack of documentation does not affect the quality of the Unit 3 PSA nor does it indicate an inadequacy of the model for use in the application of the Risk-Informed ISI program.

Attachment

BFN Unit 3 Revised RI-ISI PAGES

The revised BFN Unit 3 RI-ISI pages are:

Page E-10
Page E-19
Page E-28
Page E-29 (No revision, roll-over page)
Page E-29a
Page E-29b
Page E-30

3.2 Segment Definition

Once the systems to be included in the program are determined, the portions of the selected systems to be evaluated are divided into segments. A piping segment is defined as a run of piping whose failure would result in the same loss of function, as determined from the plant PSA or other considerations (functions which do not impact CDF). In addition, consideration was given to identifying distinct segment boundaries at branching points such as flow splits or flow joining points, locations of size changes, isolation valve, motor operated valve (MOV) and air operated valve (AOV) locations. The number of segments identified per system is given in Table 3.1-1. Description of each system's individual segments is provided in that system's section of Appendix A to 3-SI-4.6.G.

Table 3.1-1
Systems in Risk-Informed Inservice Inspection Scope

Syst		Sec XI	PSA	Mnt Rule risk significant	# Segs
001	Main Steam	Yes	Yes	Yes	56
002	Condensate and Demineralized Water Portions which provide a heat sink, or provide water to mitigate accidents, or deliver water to FW		Yes	Yes	36
003	Feedwater	Yes	Yes	Yes	46
023	Residual Heat Removal Service Water	Yes	Yes	Yes	45
024	Raw Cooling Water	Yes	Yes	Yes	20
027	Condenser Circulating Water Portion which provides cooling water to main condenser		Yes	Yes	3
063	Standby Liquid Control	Yes	Yes	Yes	5
067	Emergency Equipment Cooling Water	Yes	Yes	Yes	28
068	Reactor Recirculation	Yes	Yes		16
069	Reactor Water Cleanup	Yes	Yes		19
070	Reactor Building Closed Cooling Water	Yes	Yes		17
071	Reactor Core Isolation Cooling	Yes	Yes	Yes	12
073	High Pressure Coolant Injection	Yes	Yes	Yes	11
074	Residual Heat Removal	Yes	Yes	Yes	31
075	Core Spray	Yes	Yes	Yes	15
078	Fuel Pool Cooling	Yes			1
085	Control Rod Drive Hydraulics	Yes	Yes	Yes	31

For defense in depth all additional segments with RRW > 1.001 and those segments which could result in a large LOCA (initiating events LLC, LLD, LLO, or LLS) are considered for examination. These segments are shown in Table 3.7-2. With the addition of these segments, 97.75% of total core damage frequency due to pipe failures is accounted for.

Table 3.7-2
Defense in Depth Segments

Segment	Description	Segment CDF	%Applicable CDF	Cum % CDF	RRW
3-068-016	28" suction line from Reactor (N1B) to Recirculation pump "B"	2.83E-08	0.24%	97.28%	1.002
3-068-002	28" discharge line from Recirculation pump "A" to Recirc ring header	2.68E-08	0.23%	97.51%	1.002
3-068-004	12" discharge line from Recirc ring header "A" to Reactor (N2F)	1.52E-08	0.13%	97.64%	1.001
3-068-009	12" discharge line from Recirc ring header "B" to Reactor (N2E)	7.48E-09	0.06%	97.70%	1.000
3-068-014	28" discharge line from Recirculation pump "B" to Recirc ring header	3.20E-09	0.03%	97.73%	1.000
3-073-001	10" supply line from 26" MS line "B" to penetration X-11	2.82E-09	0.02%	97.75%	1.000
3-068-003	22" line Recirc ring header "A"	0.00E+00	0.00%	97.75%	1.000
3-068-015	22" line Recirc ring header "B"	0.00E+00	0.00%	97.75%	1.000

Large Early Release Frequency (LERF) was also considered in determining segment significance. All segments with a LERF RRW >1.001 were already selected for examination based on CDF RRW.

The contribution of each system to CDF and to LERF was calculated and was shown in Table 3.5-1. The predominant contributors to CDF are Core Spray, Reactor Recirculation, Residual Heat Removal, and Reactor Water Clean Up with Feedwater and Main Steam also contributing. The same systems also contribute to LERF. The significance of all of these systems is due to the possibility of a large LOCA, in combination with active degradation mechanisms (FAC and IGSCC).

Table 3.7-3 shows the distribution of system segments by both consequence and risk categories, along with the final designation as High Safety Significant by the Expert Panel. All of the segments which contribute to the risk distribution described above were selected by the Expert Panel. Since the Expert Panel decided to include all Medium Risk Category segments, no further re-consideration was needed.

Table 3.10-2 provides a comparison of CDF/LERF for the current and Risk-Informed programs. CDF under the current Section XI program is 99.88 percent of the base CDF and base LERF with no ISI. Under the current Augmented programs, CDF is 95.17 percent of base and LERF is 95.20 percent of base. For the Risk-Informed program, CDF is 91.72 percent of base and LERF is 91.76 percent of base, representing a reduction in risk. This reduction reflects selection of welds for inspection which have higher probabilities of CDF contributions. Current methods include selection of a random percentage of welds, which may not select those which contribute to CDF.

Table 3.10-2
COMPARISON OF APPLICABLE CDF/LERF FOR CURRENT PROGRAMS
AND FOR RISK-INFORMED PROGRAM

Program	Piping CDF	Piping LERF
Without ISI	1.159E-05	3.251E-06
Current Section XI	1.157E-05 (99.88%)	3.247E-06 (99.88%)
Current Augmented	1.103E-05 (95.17%)	3.095E-06 (95.20%)
Current Section XI +Augmented	1.101E-05 (95.00%)	3.091E-06 (95.08%)
Risk-Informed	1.063E-05 (91.72%)	2.983E-06 (91.76%)

The change in CDF due to the Risk-Informed ISI program is a reduction of 9.6E-07 with respect to the base CDF, a reduction of 9.4E-07 with respect to the current section XI, a reduction of 4.0E-07 with respect to the current augmented programs, and a reduction of 3.8E-07 with respect to the total current programs including both Section XI and augmented. The change in LERF due to the Risk-Informed ISI program is a reduction of 2.68E-07 with respect to the base LERF, a reduction of 2.64E-07 with respect to the current section XI, a reduction of 1.12E-07 with respect to the current augmented programs, and a reduction of 1.08E-07 with respect to the total current programs including both Section XI and augmented.

Defense-In-Depth

The basic concept of defense-in-depth is to provide multiple means to accomplish safety functions and prevent the release of radioactive materials.

Multiple means to accomplish safety functions are provided by the functional redundancy inherent in plant design. The PSA used as the basis of this analysis models these redundant functions. Individual quantifications were performed in this PSA for each instance in which a potential pipe failure impacted a mitigating system with no specific associated initiating event. These quantifications incorporated all potential initiating events, maintaining the system redundancy inherent to maintaining defense-in-depth.

Defense-in-depth with respect to radioactive material is maintained by assuring there are multiple barriers to release. The first barrier is the fuel cladding, whose damage is the basis for the Core Damage Frequency metric basic to this analysis. The next barrier is reactor coolant pressure boundary integrity. To assure that this barrier is maintained, additional areas are identified for their contribution to reducing risk of core damage frequency. Specifically, piping which could potentially result in a large LOCA was included, even if the risk associated with the segment was minimal or nonexistent. Additionally, reactor coolant pressure boundary integrity is maintained by continued implementation of pressure testing and visual examination per ASME Section XI.

4. IMPLEMENTATION AND MONITORING PROGRAM

A proposed revision to TVA BFN Surveillance Instruction 3-SI-4.6.G has been written to implement and monitor the RI-ISI Program. That revision complies with the guidelines described in Regulatory Guide 1.174 and 1.178 (Trial Use) and implemented in the ASME Boiler and Pressure Vessel Code, Section XI, as Code Case N-577. Upon approval of the RI-ISI program, that revision will be implemented. The new program will be integrated into the existing ASME Section XI interval. No changes to the Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the Code not affected by this change will be retained, such as inspection methods, acceptance guidelines, pressure testing, corrective measures, documentation requirements, and quality control requirements. Existing ASME Section XI program implementing procedures will be retained and modified to address the RI-ISI process, as appropriate. Additionally, the procedures include the high safety significant locations in the program requirements regardless of their current ASME class.

The proposed monitoring and corrective action program will contain the following elements:

A. Identify
B. Characterize
C. (1) Evaluate, determine the cause and extent of the condition identified
(2) Evaluate, develop a corrective action plan or plans
D. Decide
E. Implement
F. Monitor
G. Trend

The RI-ISI program is a living program requiring feedback of new relevant information to ensure the appropriate identification of high safety significant piping locations. As a minimum risk ranking of piping segments will be reviewed and adjusted on an ASME period basis. Significant changes may require more frequent adjustment as directed by NRC bulletin or Generic Letter requirements, or by plant specific feedback.

Program Implementation

The RI-ISI program will be maintained and adjusted for new relevant information. The control process to adjust the RI-ISI program will include the following inputs:

- Changes to plant design features.
- Changes to plant procedures and PSA.
- Equipment performance changes.
- Information on individual plant and industry failures.
- Examination results

During each operating cycle, the Program Owner will maintain an awareness of input changes. The BFN site control processes that provide input into the RI-ISI program will be enhanced to include the appropriate guidance. After each refueling outage, the effects of the changes will be evaluated to determine if a change to the Program is required.

The RI-ISI program will be updated, if required, before the next refueling outage. The Maintenance Rule Expert Panel will review proposed RI-ISI program changes and provide program oversight. The following provides an overview of the RI-ISI program inputs.

Changes to Plant Design Features

Design changes have the potential to change piping configuration and alter stress calculations which were used as input to the calculations performed in support of the RI-ISI program. New systems and branch piping will be evaluated for inclusion into the scope of the RI-ISI program. Consequently, the Design Control program will be revised to recognize RI-ISI and to ensure impact is appropriately evaluated during design preparation, review, and implementation. The existing design impact review process will also be used to ensure the impact of design changes on RI-ISI has been appropriately considered prior to final approval. The calculations supporting the RI-ISI program will be entered into TVA's calculation tracking program to ensure appropriate predecessors and inputs are identified and considered during design change preparation and review.

Changes to PSA

Since the PSA forms the basis for the RI-ISI program, any changes to the PSA or risk significance determination will be evaluated for impact on the RI-ISI program. This would also include changes to risk significance categories mandated by the Maintenance Rule Expert Panel. The PSA and Design Control procedures will be revised to ensure PSA changes also consider changes to the RI-ISI program and that RI-ISI changes are initiated as required.

Changes to Plant Procedures

Changes to plant procedures that affect ISI, such as system operating parameters, test interval, or the ability of plant operations to perform actions associated with accident mitigation will be evaluated for effect on the program.

Equipment Performance Changes

Equipment performance changes will be reviewed with appropriate plant personnel (e.g., system engineers, maintenance etc.,) to ensure that changes in performance parameters (e.g., valve leakage, increased pump testing, vibration problems) are considered in the RI-ISI update. Adverse equipment performance will be evaluated for changes to the RI-ISI inspection scope.

Information on Individual Plant and Industry failures

The Program Owner will consider applicable piping failures or degradations identified by the site's corrective action program. Industry awareness will be maintained through the sites Operating Experience program, NRC Generic Letters and Bulletins, site participation in Boiling Water Owners Group initiatives, and participation in the ASME Section XI Code committee activities.

Examination Results

NDE examinations, pressure tests, and corresponding VT-2 visual examinations for leakage that are determined to have unacceptable flaws, evidence of service related degradation or indications of leakage will be evaluated for effect to the program.

RI-ISI Program Review

The Maintenance Rule Expert Panel will provide the oversight role for the RI-ISI program. The Expert Panel will review proposed changes to the program.

As with past reviews, personnel possessing expertise in RI-ISI evaluation and ISI inspection/evaluation will be present during presentation and review of the above items.

5. PROPOSED ISI PROGRAM PLAN CHANGE

The locations selected for examination in the RI-ISI program and augmented programs were compared to the locations examined under the previous programs. The results are tabulated in Table 5-1. The current ASME Section XI selects a total of 222 locations for non-destructive exams, while the proposed RI-ISI program selects 70 locations for exams and credits 15 FAC segments, which results in a reduction of 152 non-destructive exam locations (68.5%). The current Generic Letter 88-01 augmented program for IGSCC selects a total of 164 locations for non-destructive exams while the proposed RI program selects 137 locations for exams, which results in a reduction of 27 non-destructive exam locations (16.5%).

Table 5-1
STRUCTURAL ELEMENT SELECTION RESULTS AND COMPARISON TO ASME SECTION XI 1989 EDITION REQUIREMENTS AND GL 88-01 REQUIREMENTS

		Current											Proposed (a) (b) (c)							
		ASME XI Elements (d)				Augmented Elements							RI-ISI Examinations							
System	# Segs	B-F	B-J	C-F-1	C-F-2	A	C	D	E	G	Dual Credit (XI & Aug)	FAC(e)	R1.11	R1.16	R1.18	A	C	D	E	G
001 MS	56		38		10							295			4					
002 CDW	36											478								
003 FW	46		23									321			11					
023 RHRSW	45																			
024 RCW	20																			
027 CCW	3																			
063 SLC	5																			
067 EECW	28																			
068 RECIRC	16	14	18			44	32		9		32 Cl 1			28 A 9 E		28	32		9	
069 RWCU	19		7			19	1				6 Cl 1			8 A		8	1			
070 RBCCW	17																			
071 RCIC	12		1		5															
073 HPCI	11		5	5	11								1							
074 RHR	31		10	2	35	4	27	2	1	2	10 Cl 1 2 Cl 2			7 C 1 E 2 G			27	2	1	2
075 CS	15	2	10	6	13		19				10 Cl 1			4 A 10 C		4	19			
078 FPC	1																			
085 CRD	31	1			6		4				1 Cl 1						4			
Total Examinations	392	17	112	13	80	67	83	2	10	2	59 Cl 1 2 Cl 2		1	69	15	40	83	2	10	2
Total Elements	1383	17	357	111	898															

Notes: (a) System pressure test requirements and VT-2 visual examinations shall continue to be performed in all ASME Code Class 1, 2, and 3 systems.

(b) Augmented programs including FAC and Reactor Nozzle Thermal Fatigue Cracking (NUREG-0619) continue

(c) Augmented program for IGSCC Categories C through G (GL88-01, NUREG-0313) continues.

(d) The current ASME Section XI ISI Program examines a minimum of 25% of the Class 1 and a minimum of 7.5% of the Class 2 elements

(e) The FAC Augmented Program examines approximately 10% of the identified locations each refueling outage.