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SOUTHERN COMPANY
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Washington, D. C. 20555

VOGTLE ELECTRIC GENERATING PLANT
RISK-INFORMED INSERVICE INSPECTION PROGRAM -
ASME CODE CATEGORY B-F, B-J, C-F-1, AND C-F-2 PIPING

Ladies and Gentlemen:

As required by 10 CFR 50.55a, the Vogtle Electric Generating Plant (VEGP) Units 1 and 2 Inservice Inspection (ISI) Program for Class 1, 2, and 3 components is based on the 1989 Edition of Section XI to the ASME Boiler and Pressure Vessel Code. In accordance with 10 CFR 50.55a(a)(3), Southern Nuclear Operating Company (SNC) requests to use the enclosed Risk-Informed Inservice Inspection (RI-ISI) Program as an alternative to the VEGP Units 1 and 2 ISI Program requirements for ASME Code Category B-F, B-J, C-F-1, and C-F-2 piping only. The proposed alternative is based on the risk-informed process described in Westinghouse Owners Group (WOG) WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," and WCAP-14572, Revision 1-NP-A, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection."

The enclosed RI-ISI Program supports the conclusion that the proposed alternative provides an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i). This program also meets the intent and principles of NRC Regulatory Guides 1.174 and 1.178. Implementation of this RI-ISI Program should allow SNC to realize significant savings in cost and radiation exposure (i.e., > $1,000,000) during the remaining life of the plant.

Southern Nuclear Operating Company requests NRC approval of the VEGP RI-ISI Program by March 31, 2003, in order to support implementation of the Program during the VEGP Unit 1 1R11 Maintenance/Refueling Outage currently scheduled to begin in the fall of 2003.

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Should there be any questions, please contact this office.

Sincerely,

J. B. Beasley, Jr.

JBB/DRG/

Enclosure: Vogtle Electric Generating Plant Units 1 and 2 Risk-Informed Inservice Inspection (RI-ISI) Program Submittal Using WOG Methodology (WCAP-14572, Rev.1) (Revision 1 Template)

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# VOGTLE ELECTRIC GENERATING PLANT
# UNITS 1 AND 2

## RISK-INFORMED INSERVICE INSPECTION (RI-ISI)
## PROGRAM SUBMITTAL
Using WOG Methodology (WCAP-14572, Rev. 1)
(Revision 1 Template)

Rev. 0

# RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN

## Table of Contents


1. INTRODUCTION/RELATION TO NRC REGULATORY GUIDE 1.174

   1.1 Introduction
   1.2 PRA Quality

2. PROPOSED ALTERNATIVE TO CURRENT INSERVICE INSPECTION PROGRAMS

   2.1 ASME Section XI
   2.2 Augmented Programs

3. RISK-INFORMED ISI PROCESS

   3.1 Scope of Program
   3.2 Segment Definitions
   3.3 Consequence Evaluation
   3.4 Failure Assessment
   3.5 Risk Evaluation
   3.6 Expert Panel Categorization
   3.7 Identification of High Safety Significant Segments
   3.8 Structural Element and NDE Selection
   3.9 Program Relief Requests
   3.10 Change in Risk

4. IMPLEMENTATION AND MONITORING PROGRAM

5. PROPOSED ISI PROGRAM PLAN CHANGE

6. SUMMARY OF RESULTS AND CONCLUSIONS

7. REFERENCES/DOCUMENTATION

# 1. INTRODUCTION/RELATION TO NRC REGULATORY GUIDE-1.174

## 1.1 Introduction

Inservice inspections (ISI) are currently performed on Vogtle Electric Generating Plant Units 1 and 2 (VEGP-1 and VEGP-2, respectively) piping to the requirements of the 1989 Edition of the Section XI to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, as required by 10CFR50.55a. Both units are currently in the second 10-year inspection interval as defined by the Code for Program B.

The objective of this submittal to the NRC is for Southern Nuclear Operating Company (SNC) to request a change to the VEGP ISI program for piping through the use of a risk-informed ISI program. The risk-informed process used in this submittal is described in Westinghouse Owners Group WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," and WCAP-14572, Revision 1-NP-A, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," (referred to as "WCAP-14572, A-Version" for the remainder of this document). "

As a risk-informed application, this submittal meets the intent and principles of NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis". Further information is provided in Section 3.10 relative to defense-in-depth.

## 1.2 PRA Quality

The Plant Vogtle-specific Level 1 and Level 2 Probabilistic Risk Assessment (PRA) Model, Revision 2C, was used to evaluate the impacts on plant risk of pipe ruptures during power operation. This model, when used in conjunction with deterministic evaluations, is of sufficient quality to support regulatory applications such as this submittal, as described below. The associated PRA calculations performed as part of the development of this RI-ISI submittal were originated, verified, approved and documented in accordance with SNC procedures for the preparation and control of calculations.

As an integral part of the its initial development pursuant to NRC Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities", the PRA was repeatedly reviewed by an Independent Review Group which included experts in plant design, plant operation, and probabilistic risk assessment. Further, each subsequent revision to the model has been internally reviewed and approved in accordance with applicable SNC procedures. In addition, an evaluation based upon Appendix B of the EPRI PSA Applications Guide was performed to confirm that the PRA conforms to the industry state-of-the-art practices with respect to the scope of potential plant scenarios.

In December 2001, the PRA was extensively reviewed by an experienced five-man Peer Review Team coordinated by the Westinghouse Owners Group in a manner described in the Nuclear Energy Institute's document NEI 00-02, "Industry Peer Review Process". (Note that the peer

review report is currently in draft form pending final issuance by Westinghouse). The peer review evaluated the eleven elements of the PRA and concluded that all elements were either a "Grade 3" or a "Contingency Grade 3". A "Grade 3" is defined in the Peer Review Process as:

> "This grade extends the requirements *[of previously defined Grades 1 and 2]* to assure that the risk significance determinations made by the PRA are adequate to support regulatory applications, when combined with deterministic insights. Therefore, a PRA with elements determined to be at Grade 3 can support physical plant changes when it is used in conjunction with other deterministic approaches that ensure that defense-in-depth is preserved. Grade 3 is acceptable for Grade 1 and 2 applications, and also for assessing safety significance of equipment and operator actions. This assessment can be used in licensing submittals to the NRC to support positions regarding absolute levels of safety significance if supported by deterministic evaluations."

Eight PRA elements were judged by the peer review to have findings that resulted in their being considered "Contingency Grade 3". A "Contingency Grade 3" reverts to a "Grade 3" when items noted in the evaluation of the element are resolved. Such pending items are classified as one of four degrees of significance. None of the pending items noted in the Plant Vogtle PRA evaluation were judged to be of a level of significance to require prompt resolution to ensure the technical adequacy of the PRA. Therefore, even though considered important for the long-term enhancement of the PRA model, the resolution of the subject items has been deferred until the next periodic PRA model update. This deferral does not adversely affect the use of the current Revision 2C of the PRA model for applications such as the RI-ISI when supplemented by deterministic evaluations.

The base Core Damage Frequency (CDF) and the base Large Early Release Frequency (LERF) from this revision of the PRA model are 1.60E-05/rx-yr and 7.65E-08/rx-yr, respectively.

## 2. PROPOSED ALTERNATIVE TO ISI PROGRAM

### 2.1 ASME Section XI

Examination Categories B-F, B-J, C-F-1, and C-F-2 of the ASME Section XI Code currently contain the scope and non-destructive examination requirements for piping components. These current requirements are limited to ASME Class 1 and Class 2 piping with specific size and pressure/temperature exemptions. The alternative RI-ISI program is described in WCAP-14572, A-Version. It is a substitute for the current examination program on piping in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety.

In addition, the alternative program is not limited to the current examination scope in ASME Class 1 or Class 2 piping but encompasses all of the Class 1 and Class 2 high safety significant piping segments, regardless of previous Code exemptions. Other non-related portions of the ASME Section XI Code are unaffected. WCAP-14572, A-Version, provides the requirements, which defines the relationship between the risk-informed examination program and the remaining, unaffected portions of ASME Section XI.

## 2.2 Augmented Programs

Augmented weld inspection programs include weld examinations in high-energy Main Steam and Feedwater systems outside containment. Flow-accelerated corrosion (FAC) examinations for wall-thinning are conducted in Feedwater, Auxiliary Feedwater, and Steam Generator Blowdown systems. The implementation of this Class 1 and 2 RI-ISI program has no effect on these augmented inspection programs.

# 3. RISK-INFORMED ISI PROCESSES

The processes used to develop the RI-ISI program are consistent with the methodology described in WCAP-14572, A-Version.

The process that is being applied, involves the following steps:

- Scope Definition,
- Segment Definition,
- Consequence Evaluation,
- Failure Assessment,
- Risk Evaluation,
- Expert Panel Categorization,
- Element/NDE Selection,
- Implement Program, and
- Feedback Loop.

Deviations

Credit for Leak Detection was used only for the Reactor Coolant System (RCS) as prescribed by WCAP-14572, A-Version, except that two additional segments in the Nuclear Sampling (NS) System were credited for Leak Detection. A break in either of these two NS System segments is assumed to cause a small break Loss of Coolant Accident (LOCA) inside containment; therefore, it is appropriate to use RCS leak detection.

As part of the risk evaluation described in Section 3.5, the uncertainty analysis as described on page 125 of WCAP-14572, A-Version was performed and is now included as part of the base process.

## 3.1 Scope of Program

All VEGP Class 1 and 2 pressure-retaining piping is included in the RI-ISI program. The applicable systems are listed in Tables 3.1-1 and 3.1-2 for VEGP-1 and VEGP-2, respectively.

## 3.2 Segment Definitions

Once the systems to be included in the program were determined, the piping for these systems was divided into segments.

The number of pipe segments defined for the 11 systems are summarized in Tables 3.1-1 and 3.1-2 for VEGP-1 and VEGP-2, respectively.

The Piping and Instrumentation Diagrams were used to define the segments.

## 3.3 Consequence Evaluation

The consequences of pressure boundary failures are measured in terms of core damage and large early release. The impact on these measures due to both direct and indirect effects was considered. Table 3.3-1 summarizes the postulated consequences for each system, including both the direct and indirect effects.

## 3.4 Failure Assessment

Failure estimates were generated utilizing industry failure history, plant specific failure history, and other relevant information. An engineering team was established that had access to expertise in the following areas: ISI, NDE, materials, stress analysis, and system engineering. The team was trained in the failure probability assessment methodology and the Westinghouse SRRA code, including identification of the capabilities and limitations as described in WCAP-14572, A-Version, Supplement 1. The SRRA code was 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 Supplement 1 of the WCAP. The engineering team assessed industry and plant experience, plant layout, materials, and operating conditions and identified the potential failure mechanisms and causes as input into the SRRA model. All the piping configurations included in the RI-ISI program were adequately modeled using the SRRA code.

As a bench-mark, the SRRA code was used for calculating failure probabilities for Intergranular Stress Corrosion Cracking (IGSCC) of Boiling Water Reactor plant piping, as described in WCAP-14572, A-Version, Supplement 1. A range of SRRA input values was determined for VEGP-1 and VEGP-2 based on factors such as material type (e.g., Inconel, 304SS, 304L SS), temperatures, and oxygen content. The results were compared with plant and industry failure data (including recent events) to ensure that the selected input values were reasonable.

Sensitivity studies were performed to aid in determining representative input values when sufficient information was not available. Snubber failure history was also reviewed to identify any potential effects that could increase piping failure probability.

Table 3.4-1 summarizes the failure probability estimates for the dominant potential failure mechanism(s)/combination(s) by system for both VEGP units. Table 3.4-1 also describes the dominant failure mechanism and its location(s) within the system for both units.

Another consideration was whether a segment was addressed by either augmented weld examinations or by augmented programs designed to detect FAC. For augmented weld examinations, the effects of ISI were included in the risk evaluations and were used to assist in categorizing the segments as described on page 105 of WCAP-14572, A-Version. For FAC, the EPRI CHECWORKS™ program along with plant-specific FAC wall-thinning monitoring program data were used as input for the SRRA calculations. Credit was taken for detecting wall thinning and replacing degraded pipe prior to its failure. Where credit for FAC mitigation was included in

the program inputs, the program output without ISI was selected to be used in downstream calculations.

The failure probabilities used in the risk-informed process are documented and maintained in the plant records

## 3.5 Risk Evaluation

Each piping segment within the scope of the program was evaluated to determine the CDF and LERF, resulting from the postulated piping failure. Calculations were performed considering cases with and without operator action.

Once this evaluation was completed, the total pressure boundary CDF and LERF were calculated by summing across the segments for each system.

The results of these calculations are presented in Table 3.5-1 for VEGP-1. The CDF due to piping failure without operator action is 8.30E-07/year, and with operator action is 7.47E-07/year. The total LERF due to piping failure without operator action is 4.78E-09/year, and with operator action is 4.17E-09/year.

The results of these calculations are presented in Table 3.5-2 for VEGP-2. The total core damage frequency due to piping failure without operator action is 8.32E-07/year, and with operator action is 7.48E-07/year. The total large early release frequency due to piping failure without operator action is 4.79E-09/year, and with operator action is 4.17E-09/year.

The uncertainty analysis, as described on WCAP page 125, was performed and is now included as part of the base process.

To assess safety significance, the risk reduction worth (RRW) and risk achievement worth (RAW) were calculated for each piping segment.

## 3.6 Expert Panel Categorization

The final safety determination (i.e., high and low safety significance) of each piping segment was made by the expert panel using both probabilistic and deterministic insights. The expert panel was comprised of personnel with expertise in the following fields: probabilistic safety assessment, plant operations, plant and industry maintenance, repair and failure history, and system design and operation. Members associated with the Maintenance Rule were used to ensure consistency with the other PRA applications. Alternates with similar expertise and training as the regular expert panel members were used, as necessary.

The expert panel core team was composed of at least five members and their alternates from the following functional areas:

1. Operations (Senior Reactor Operator),
2. Maintenance,
3. Engineering,
4. Outage and Modifications, and
5. Nuclear Safety and Compliance.

A minimum of 4 members or alternates filling the above positions constituted a quorum, with no more than two alternates participating in the voting. This core team was supplemented, as necessary, by other experts. Available supplemental expertise included system engineers, stress engineers, ISI engineers, NDE personnel, PRA personnel, and personnel knowledgeable of SRRA methods (including uncertainty).

An expert panel chairperson was appointed by the Engineering Support Manager. The chairperson conducted the expert panel meetings and participated in the voting.

Members and alternates received training in the RI-ISI selection process. They were indoctrinated in the application of risk analysis techniques for ISI. These techniques included risk importance measures, threshold values, failure probability models, failure mode assessments, PRA modeling limitations, and the use of expert judgment. Training documentation is maintained with the expert panel's records.

Worksheets were provided to the panel on each system for each piping segment containing information pertinent to the panel's selection process. This information, in conjunction with each panel member's own expertise and other documents as appropriate, was used to determine the safety significance of each piping segment.

A consensus process was used by the expert panel. Consensus was defined as a majority of Expert Panel members/alternates present with the caveat that the Engineering Support Manager would review and approve all decisions where a unanimous decision was not achieved.

The chairperson appointed a secretary to record the minutes of each meeting. The minutes included the names of members and alternates in attendance and whether a quorum was present. The minutes contained relevant discussion summaries and the results of membership voting. These minutes are available as program records.

## 3.7 Identification of High Safety Significant Segments

The number of high safety significant segments for each system, as determined by the expert panel, is shown in Table 3.7-1 and 3.7-2 for VEGP-1 and VEGP-2, respectively, along with a summary of the risk evaluation identification of high safety significant segments.

## 3.8 Structural Element and Non-Destructive Examination (NDE) Selection

The structural elements in the high safety significant piping segments were selected for examination and appropriate NDE methods were defined.

This initial program addresses the high safety significant (HSS) piping components placed in Regions 1 and 2 of Figure 3.7-1 in WCAP-14572, A-Version. Segments considered as "high failure importance" (Region 1) were identified as all segments being affected by an active failure mechanism or analyzed to be highly susceptible to a failure mechanism (probability of large leak at 40 years generally exceeds 1E-04). Regions 3 and 4 piping components, which are low safety significant, are to be considered in an Owner Defined Program and are not considered part of the program requiring approval. Region 1, 2, 3, and 4 piping components will continue to receive Code-required pressure testing, as described in the current ASME Section XI program.

For VEGP-1, 1180 piping segments were evaluated in the RI-ISI program. Region 1 contains 8 segments, Region 2 contains 73 segments, Region 3 contains 95 segments, and Region 4 contains 950 segments. (There were 54 segment numbers not used in VEGP-1 and are not included in the region counts).

The number of locations to be inspected in a HSS segment was determined using a Westinghouse statistical (Perdue) model as described in section 3.7 of WCAP-14572, A-Version. None of the 8 VEGP-1 HSS piping segments in Region 1 were evaluated using the Perdue model, because they were located completely in Region 1A, which is outside the applicability of the model. For these 8 segments, the guidance in Section 3.7.3 of WCAP-14572 A-Version was followed. All 73 of the VEGP-1 HSS piping segments in Region 2 were evaluated using the Perdue model.

For VEGP-2, 1185 piping segments were evaluated in the RI-ISI program. Region 1 contains 9 segments, Region 2 contains 72 segments, Region 3 contains 96 segments, and Region 4 contains 942 segments. (There were 66 segment numbers not used in VEGP-2 and are not included in the region counts).

One of the VEGP-2 HSS piping segments in Region 1 and all 72 of the VEGP-2 HSS piping segments in Region 2 were evaluated using the Perdue model. The 8 VEGP-2 segments that were not evaluated using the Perdue model were located completely in Region 1A, which is outside the applicability of the model. For these 8 segments, the guidance in Section 3.7.3 of WCAP-14572, A-Version was followed.

Table 4.1-1 in WCAP-14752, A-Version, was used as guidance in determining the examination requirements for the HSS piping segments. VT-2 visual examinations required by Table 4.1-1 are scheduled each refueling outage. Other VT-2 visual examinations are scheduled in

accordance with the VEGP pressure test program that remains unaffected by the RI-ISI program.

Additional Examinations

The program, in all cases, will determine through an engineering evaluation the root cause of any unacceptable flaw or relevant condition found during RI-ISI examinations. The evaluation will include the applicable service conditions and degradation mechanisms to establish that the element(s) will still perform their intended safety function during subsequent operation. Elements not meeting this requirement will be repaired or replaced.

The evaluation will determine whether there are other elements on the same segment or whether there are elements in additional segments subject to the same root cause and degradation mechanism. If so, then additional examinations will be performed on these elements, up to a number equivalent to the number of elements on the segment or segments initially required to be inspected during the outage. If unacceptable flaws or relevant conditions are again found similar to the initial problem, the remaining elements identified as susceptible will be examined. No additional examinations will be performed if there are no additional elements identified as being susceptible to the same service-related root cause conditions or degradation mechanism.

## 3.9 Program Relief Requests

Per NRC-approved Relief Request RR-1, the Second 10-Year Interval for VEGP-2 was started approximately two years ahead of the 10CFR50.55a(g)(4)(ii) ISI requirements for updating ISI programs, such that the two VEGP units now have the same 10-year interval (and inspection period) dates for the second interval. Similar relief will be pursued for the subsequent intervals. This change will continue for the RI-ISI schedules. All other existing relief requests remain in place for the second interval.

Alternate methods are specified to ensure structural integrity in cases where examination methods cannot be applied due to limitations such as inaccessibility or radiation exposure hazard.

An attempt will be made to provide a minimum of >90% coverage of the examination volume (per Code Case N-460) when performing the risk-informed examinations. However, some limitations will not be known until the examination is performed, since some locations will be examined for the first time by the specified NDE techniques. In instances, where an examination does not meet >90% coverage, a relief request will be submitted.

## 3.10 Change in Risk

The RI-ISI program for VEGP-1 and VEGP-2 has been developed in accordance with NRC Regulatory Guide 1.174, and the risk incurred from implementation of this program is slightly less than the estimated risk from current requirements.

The change-in-risk calculations were performed in accordance with the guidelines provided on page 213 of the WCAP. A comparison between the proposed RI-ISI program and the current ASME Section XI ISI program was made to evaluate the change in risk. The approach

evaluated the change in risk with the inclusion of the probability of detection as determined by the SRRA model. Adjustments were made to add segments until all four criteria for accepting the results discussed on page 214 and 215 in the WCAP were met. This evaluation resulted in the identification of 13 piping segments for VEGP-1 and 9 piping segments for VEGP-2 for which examinations are now required (systems identified in Table 5-1a for VEGP-1 and Table 5-1b for VEGP-2 via a footnote).

The results from the risk comparison are shown in Table 3.10-1 for VEGP-1 and 3.10-2 for VEGP-2. As seen from the tables, the RI-ISI program reduces the risk associated with piping CDF/LERF slightly more than the current ASME Section XI ISI program while reducing the number of examinations. Tables 3.10-1 and 3.10-2 also include the systems that are the main contributors to the risk reduction in moving from the current program to the RI-ISI program. The primary basis for this risk reduction is that examinations will be performed on high safety significant piping segments that are not currently examined in the current ASME Section XI ISI program.

Defense-In-Depth

The reactor coolant piping will continue to receive a system pressure test and VT-2 visual examinations as defined in the current ASME Section XI program document. Volumetric examinations are proposed on the smaller reactor coolant piping as part of the RI-ISI program. Larger reactor coolant loop piping segments were identified as high safety significant and will continue to be inspected and will also meet "defense-in-depth" considerations. The locations selected were associated with the reactor vessel dissimilar metal welds. Those particular locations were identified as being the area to inspect in the RI-ISI process, if the segment was chosen.

## 4. IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the RI-ISI program, SNC procedures that comply with the guidelines described in WCAP-14572, A-version, will be prepared to implement and monitor the program. The new program will be integrated into the existing ASME Section XI interval. No changes to the Updated Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the Code not affected by this change would 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 would be retained and would be modified to address the RI-ISI process, as appropriate.

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

A. Identify
B. Characterize
C. Evaluate
    1. Determine the cause and extent of the condition identified.
    2. 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 inspection period basis. Significant changes may require more frequent adjustment as directed by NRC regulatory requirements (e.g., NRC Bulletin, NRC Generic Letter, etc.), or by plant specific feedback.

## 5. PROPOSED ISI PROGRAM CHANGE

A comparison between the RI-ISI program and the current ASME Section XI program requirements for piping is given in Table 5-1a for VEGP-1 and Table 5-1b for VEGP-2. An identification of piping segments that are part of plant augmented programs is also included in Tables 5-1a and 5-1b.

For the RI-ISI program, the plant will be performing volumetric examinations on elements not currently required to be volumetrically examined by ASME Section XI. Some examples are provided below.

- The ASME Section XI Code does not require volumetric examination of Class 1 piping less than 4-inches NPS (Examination Category B-J, Item B9.21). The RI-ISI program will require volumetric examination of these welds. Examples where the risk-informed process requires volumetric examination and the Code does not require volumetric examination are the cold leg injection lines and the pressurizer power-operated relief valve lines.

- The ASME Section XI Code does not require volumetric or surface examination of Class 2 piping 4-inch nominal pipe size (NPS) or less for non-High Head Safety Injection Lines. The RI-ISI program will require volumetric examination of these welds. Examples where the risk-informed process requires examination and the Code does not are the 4" non-isolable Main Steam lines outside of containment and the 3" Chemical and Volume Control System letdown piping downstream of the letdown orifices.

- The ASME Section XI Code does not require volumetric and surface examinations of piping less than 3/8-inch wall thickness on Class 2 piping greater than 4-inch NPS. The welds are counted for percentage requirements, but not examined by NDE. The RI-ISI program will require volumetric examination of these welds. Examples where the risk-informed process requires examination and the Code does not are the 8" NPS and less suction lines from the Reactor Water Storage Tank (safety injection).

As discussed in Section 3.9, per NRC-approved Relief Request RR-1, the Second 10-Year Interval for VEGP-2 was started approximately two years early, such that the two VEGP units now have the same 10-year interval (and inspection period) dates for the second interval. The initial RI-ISI program is projected to start in second inspection period of the Second 10-Year ISI interval (both units ending the second period on May 31, 2004). Assuming approval, as projected:

- Approximately 1/3rd of the required RI-ISI examinations will be performed during the second inspection period.
- Approximately 2/3rds of the required RI-ISI examinations will be completed by the end of the second 10-year ISI interval.

## 6. SUMMARY OF RESULTS AND CONCLUSIONS

A partial scope (all Class 1 and 2 piping) RI-ISI application has been completed for VEGP-1 and VEGP-2. Upon review of the proposed risk-informed ISI examination program given in Tables 5-1a and 5-1b for VEGP-1 and VEGP-2, respectively, an appropriate number of examinations are proposed for the high safety significant segments across the Class 1 and Class 2 portions of the plant piping systems. Resources to perform examinations currently required by ASME Section XI in the Class 1 or Class 1 and Class 2 portions of the plant piping systems, even though reduced, are distributed to address the greatest amount of risk within the scope. Thus, the change in risk principle of NRC Regulatory Guide 1.174 is maintained. In addition, the examinations performed will address specific damage mechanisms postulated for the selected locations through appropriate examination selection and increased volume of examination.

Vogtle Electric Generating Plant has enhanced design features and early-in-life procedural controls using lessons learned from earlier plants, resulting in reduced failure probabilities, when compared to some earlier generation plants. Examples of enhanced features include:

- Steam generator auxiliary bypass nozzles to eliminate low-flow stratification issues.
- Selective use of 304L stainless steel and chemistry controls for halogens to reduce the potential for stress corrosion cracking in stagnant borated water areas.
- Use of an installed fatigue monitoring system to provide accurate thermal cycling information for input into the estimation of failure probability.
- Minimized early-in-life thermal cycles due to unplanned trips.
- Designed to ASME Section III for all Class 1 piping and Class 2 piping. Thus there is an improved level of fatigue analysis and operating conditions scrutiny for the ASME Section III, NB-3600 design as compared to many earlier plants.
- There is a much larger percentage of small diameter, Class 1 piping constructed with butt welds as opposed to socket welds when compared to many earlier plants. Subsequently, more detailed information is available for input to the estimation of the failure probability.

From a risk perspective, the PRA dominant accident sequences include Loss of Nuclear Service Cooling Water, Loss of Offsite Power, and Loss of Coolant Accidents.

For the RI-ISI program, appropriate sensitivity and uncertainty evaluations have been performed to address variations in piping failure probabilities and PRA consequence values along with consideration of deterministic insights to assure that all high safety significant piping segments have been identified.

As a risk-informed application, this submittal meets the intent and principles of NRC Regulatory Guide 1.174.

## 7. REFERENCES/DOCUMENTATION

### SNC RI-ISI Project References


**1) General**

a) Nuclear Services And Projects Divisions Policies & Procedures", WP-4.5 Revision 4, "Design Analysis," effective October 1, 2001.

b) WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," February 1999.

c) SAE/RRA-032 (00), "Westinghouse Electric Company LLC Project Quality Plan For Issuance Of Calculation Notes Within Reliability & Risk Assessment Pertaining To Risk-Informed Inservice Inspection Work Performed For Southern Nuclear Operating Company Vogtle Nuclear Plant," July 2000.

**2) Scope Definition**

a) SNC Calculation Note ITS-V-00-1-RI, Revision 0, Risk Informed ISI Program Scope Vogtle Units 1 & 2, June 21, 2002. (Reissue)

**3) Segment Definition**

a) SNC Letter ENG-02-002, Transmittal of Segment Definitions for AFW, CI, CV, FW, SGB, CS, NS, MS, Systems and PRA Run Mapping File (All systems), March 18, 2002, from W. E. Burns.

b) SNC Calculation Note REES-V-01-002 Revision 1, AFW Final Segment Definitions and Consequence Analysis.

c) SNC Calculation Note REES-V-01-020 Revision 1, CI Final Segment Definitions and Consequence Analysis.

d) SNC Calculation Note REES-V-01-026 Revision 1, CV Final Segment Definitions and Consequence Analysis.

e) SNC Calculation Note REES-V-01-004 Revision 1, FW Final Segment Definitions and Consequence Analysis.

f) SNC Calculation Note REES-V-01-003 Revision 1, SGB Final Segment Definitions and Consequence Analysis.

g) SNC Calculation Note REES-V-01-006 Revision 1, CS Final Segment Definitions and Consequence Analysis.

h) SNC Calculation Note REES-V-01-010 Revision 0, NS Final Segment Definitions and Consequence Analysis.

i) SNC Calculation Note REES-V-01-011 Revision 2, MS Final Segment Definitions and Consequence Analysis.

j) Westinghouse Calculation Note CN-RRA-00-32 Revision 1, Southern Nuclear Operating Company RI-ISI Segment Definition / Direct Consequence Definition for Vogtle Units 1 & 2 Residual Heat Removal System, May 2, 2002.

k) Westinghouse Calculation Note CN-RRA-00-35 Revision 1, Southern Nuclear Operating Company RI-ISI Segment Definition / Direct Consequence Definition for Vogtle Units 1 & 2 Reactor Coolant System, May 2, 2002.

l) Westinghouse Calculation Note CN-RRA-00-29 Revision 1, Southern Nuclear Operating Company RI-ISI Segment Definition / Direct Consequence Definition for Vogtle Units 1 & 2 Safety Injection System, May 2, 2002.

**4) SRRA Piping Failure Calculations**

a) SNC Calculation Note ITS-V-00-10-RI Revision 2, Vogtle Units 1 & 2 Risk-Informed SRRA Calculation Note, Auxiliary Feedwater System, March
b) 20, 2002.
c) SNC Calculation Note ITS-V-00-18-RI Revision 5, Vogtle Units 1 & 2 Risk-Informed SRRA Calculation Note, Containment Penetration System, July 3, 2002.
d) SNC Calculation Note ITS-V-00-8-RI Revision 2, Vogtle Units 1 & 2 Risk-Informed SRRA Calculation Note, Containment Spray System, March 20, 2002.
e) SNC Calculation Note ITS-V-00-19-RI Revision 5, Vogtle Units 1 & 2 Risk-Informed SRRA Calculation Note, Chemical and Volume Control System, July 3, 2002.
f) SNC Calculation Note ITS-V-00-14-RI Revision 2, Vogtle Units 1 & 2 Risk-Informed SRRA Calculation Note, Feedwater System, March 20, 2002.
g) SNC Calculation Note ITS-V-00-15-RI Revision 2, Vogtle Units 1 & 2 Risk-Informed SRRA Calculation Note, Main Steam System, March 20, 2002.
h) SNC Calculation Note ITS-V-00-17-RI Revision 3, Vogtle Units 1 & 2 Risk-Informed SRRA Calculation Note, Nuclear Sampling System, May 8, 2002.
i) SNC Calculation Note ITS-V-00-13-RI Revision 4, Vogtle Units 1 & 2 Risk-Informed SRRA Calculation Note, Reactor Coolant System, July 3, 2002.
j) SNC Calculation Note ITS-V-00-11-RI Revision 4, Vogtle Units 1 & 2 Risk-Informed SRRA Calculation Note, Residual Heat Removal System, July 3, 2002.
k) SNC Calculation Note ITS-V-00-16-RI Revision 2, Vogtle Units 1 & 2 Risk-Informed SRRA Calculation Note, Steam Generator Blowdown System, March 21, 2002.
l) SNC Calculation Note ITS-V-00-9-RI Revision 4, Vogtle Units 1 & 2 Risk-Informed SRRA Calculation Note, Safety Injection System, July 3, 2002.

**5) Indirect Effects**

a) SNC Calculation Note ITS-V-00-12-RI Revision 0, Vogtle Units 1 & 2 Risk-Informed Indirect Effects Calculation Note, February 18, 2002.
b) Westinghouse Letter LTR-RRA-02-83, Vogtle Units 1 & 2 Risk Informed ISI Plant Walkthrough Notebook, June 25, 2002.
c) Westinghouse Letter; SAE/RRA-077 (00), Transmittal of SNC RI-ISI Plant Walk-through Notes, September 26, 2000.

**6) PRA Analyses**

a) SNC Letter ENG-02-004, Vogtle Unit 1 & 2 Revised Feedwater PRA Run Values, April 2, 2002.
b) SNC Letter ENG-02-002, Vogtle Unit 1 & 2 Final Segment Definition and Consequence Analysis and Final Segment to PRA Run Number mapping File, March 18, 2002.
c) SNC Calculation Note REES-V-01-005 Revision 1, Final Consolidated Application Table.
d) SNC Letter ENG-02-006, Vogtle Unit 1 & 2 Changes to PRA Runs CS22 and RH52, May 24, 2002.
e) SNC Letter ENG-02-003, Vogtle Unit 1 & 2 PRA Run Values, March 25, 2002.

7) **Expert Panel**
   a) Westinghouse Letter LTR-RRA-02-69, Vogtle Units 1 & 2 Expert Panel Workbook Listing, N. B. Closky Notes, June 7, 2002.
   b) Westinghouse Letter LTR-RRA-02-13, Vogtle Units 1 & 2 Expert Panel Workbook Listing, P. R. Stevenson Notes, January 29, 2002.
   c) SNC Letter ITS02VNP025, Transmittal of Expert Panel Minutes, March 20, 2002.
   d) Westinghouse Letter LTR-RRA-02-20, Vogtle Units 1 & 2 Changes to RI-ISI Program from 1st Expert Panel to 2/14/02.
   e) Westinghouse Letter LTR-RRA-02-71, Vogtle Units 1 & 2 Expert Panel #2 Review Spreadsheet, June 7, 2002.
   f) SNC Letter ITS02VNP054, Expert Panel #3, Meeting Minutes, June 7, 2002.
   g) Westinghouse Calculation Note CN-RRA-02-49, Vogtle Units 1 & 2 Risk-Informed Inservice Inspection Expert Panel Database, July 2002.

8) **Risk Ranking**
   a) Westinghouse Letter LTR-RRA-02-11, Vogtle Units 1 & 2 RI-ISI Risk Ranking Spreadsheet, Expert Panel #1; October 21, 2001 through October 26, 2001.
   b) Westinghouse Letter LTR-RRA-02-12, Vogtle Units 1 & 2 RI-ISI Risk Ranking Spreadsheet, Expert Panel #2, December 14, 1001
   c) SNC Letter ITS02VNP030, Transmittal of Gas Containing Segments, March 28, 2002
   d) Westinghouse Calculation Note CN-RRA-02-23 Revision 0, Vogtle Units 1 & 2, Risk-Informed Inspection Risk Ranking, July 2002.

9) **Change in Risk**
   **a)** SNC Letter ITS02VNP033, Transmittal of Revised Vogtle HSS/LSS Segments, April 9, 2002.
   **b)** Westinghouse Calculation Note CN-RRA-02-43, Revision 0, Southern Nuclear RI-ISI Delta Risk for Vogtle 1 and Vogtle 2, July 2002.

10) **Perdue Modeling**
   a) SNC Letter ITS02VNP040, Transmittal of Revised HSS Segment Weld Information, April 29, 2002.
   b) SNC Letter ITS02VNP037, Transmittal of RI-ISI Segments in Flow Accelerated Corrosion (FAC) Program, April 24, 2002.
   c) Westinghouse Calculation Note CN-RRA-02-40 Revision 0, Southern Nuclear Operating Company RI-Perdue Modeling for Vogtle Units 1 & 2, July 1, 2002.

11) **Final Report**
   a) Westinghouse Project Letter GP-17400, Southern Nuclear Operating Company Vogtle Electric Generating Plant Units 1 and 2 Risk-Informed Inspection Draft Submittal, May 31, 2002.
   b) Westinghouse Letter LTR-RRA-02-87, Transmittal of Final Vogtle Units 1 & 2 RI-ISI Draft Submittal Comments, July 2, 2002.

| Table 3.1-1<br>System Selection and Segment Definition for<br>VEGP-1 Class 1 and 2 Piping | | | |
|---|---|---|---|
| **System Description** | **PRA** | **Section XI** | **Number of Segments** |
| Auxiliary Feedwater System (AFW) | Yes | Yes | 48 |
| Containment Isolation System (CI) | Yes | No | 169 |
| Containment Spray System (CS) | Yes | Yes | 66 |
| Chemical and Volume Control System (CV) | Yes | Yes | 274 |
| Feedwater System (FW) | Yes | Yes | 34 |
| Main Steam System (MS) | Yes | Yes | 80 |
| Nuclear Sampling System (NS) | Yes | No | 9 |
| Reactor Coolant System (RCS) | Yes | Yes | 172 |
| Residual Heat Removal System (RHR) | Yes | Yes | 98 |
| Steam Generator Blowdown System (SGB) | Yes | No | 80 |
| Safety Injection System (SI) | Yes | Yes | 150 |
| Total | | | 1180 |

| Table 3.1-2<br>System Selection and Segment Definition for<br>VEGP-2 Class 1 and 2 Piping | | | |
|---|---|---|---|
| **System Description** | **PRA** | **Section XI** | **Number of Segments** |
| Auxiliary Feedwater System (AFW) | Yes | Yes | 48 |
| Containment Isolation System (CI) | Yes | No | 169 |
| Containment Spray System (CS) | Yes | Yes | 66 |
| Chemical and Volume Control System (CV) | Yes | Yes | 274 |
| Feedwater System (FW) | Yes | Yes | 34 |
| Main Steam System (MS) | Yes | Yes | 80 |
| Nuclear Sampling System (NS) | Yes | No | 9 |
| Reactor Coolant System (RCS) | Yes | Yes | 172 |
| Residual Heat Removal System (RHR) | Yes | Yes | 103 |
| Steam Generator Blowdown System (SGB) | Yes | No | 80 |
| Safety Injection System (SI) | Yes | Yes | 150 |
| Total | | | 1185 |

| Table 3.3-1<br>Summary of Postulated Consequences by System for<br>VEGP-1 and VEGP-2 Piping | |
|---|---|
| ***System Description*** | **Summary of Consequences** |
| AFW - Auxiliary Feedwater | Segment failure leads to a reactor trip or Secondary-Side line break initiating events. Mitigating system impacts include loss of AFW pump, Main Feedwater (MFW) pump, Condensate Pump and containment isolation functions. Several segment failures increase the likelihood of containment bypass given a Steam Generator Tube Rupture (SGTR). |
| SGB - Steam Generator Blowdown | Several segment failures lead to a reactor trip or Secondary-Side line break inside containment initiating events. Mitigating system impacts include the following losses: steam supply to Turbine Driven Auxiliary Feedwater (TDAFW) pump, AFW flow to Steam Generators and Main FW/Condensate flow to Steam Generators. |
| CI - Containment Isolation | The mitigating system impact of pipe breaks in some segments is the loss of containment isolation. Failures of some pipe segments cause a reactor trip or a special initiating event (loss of all Auxiliary Component Cooling Water (ACCW) flow or loss of Instrument Air that also lead to mitigating system losses). The ACCW system provides cooling to the Reactor Coolant pump thermal barrier and to the Normal Charging pump (NCP). Loss of Instrument Air prevents the re-establishment of feedwater and/or condensate flow to the steam generators. Failures of some pipe segments cause an increase in the likelihood of a special initiating event (e.g., loss of one ACCW pump or total or partial loss of one Nuclear Service Cooling Water (NSCW) train that also leads to mitigating system losses). The NSCW system provides cooling water to the containment fan coolers, Essential Chilled Water condensers, Emergency Core Cooling pump coolers, the Diesel Generator jacket water coolers, and the CCW and ACCW heat exchangers. |
| CS - Containment Spray | Mitigating system impacts include the following: loss of CS train A or B injection, loss of sump inventory, loss of Reactor Water Storage Tank (RWST) inventory and an increased likelihood of the failure of containment isolation. |

| Table 3.3-1 (Continued)<br>Summary of Postulated Consequences by System for<br>VEGP-1 and VEGP-2 Piping | |
|---|---|
| CV - Chemical & Volume Control | Many segment failures lead to a reactor trip, small LOCA or medium LOCA initiating events. Mitigating system impacts include loss of one or two centrifugal charging pumps (CCPs), the NCP, the RWST, seal injection to one or all loops, sump inventory, emergency boration, normal charging, the Volume Control Tank (VCT) and loss of the cross-tie line to the Safety Injection Pumps (SIPs). |
| FW - Feedwater | Several segment failures lead to a reactor trip or Secondary-Side line break initiating events. Mitigating system impacts include loss of steam supply to TDAFW pump, AFW flow to Steam Generators, MFW and Condensate flow to Steam Generators. Several segment failures increase the likelihood of containment bypass given a SGTR event; several segment failures result in an immediate bypass given a SGTR event. |
| MS - Main Steam | Several segment failures lead to a reactor trip or Secondary-Side line break inside containment initiating events. Mitigating system impacts include loss of steam supply to TDAFW pump, AFW flow to Steam Generators, MFW and Condensate flow to Steam Generators and Main Steam Isolation Valves (MSIVs) to isolate. Several segment failures increase the likelihood of containment bypass given a SGTR event; several segment failures result in an immediate bypass given a SGTR event. |
| NS - Nuclear Sampling | This system is not modeled in the VEGP PSA. However, one segment interfaces with a CV segment and has the same impacts as the CV segment (i.e., impacts on emergency boration and normal charging flow paths). Two NS segments interface with RCS segments and have the same impacts as the interfacing RCS segments (i.e., reactor trip and small LOCA). |
| RCS - Reactor Coolant | Many segment failures lead to a large, medium or small LOCA, or a reactor trip initiating event. Mitigating system impacts include the following: loss of high/low pressure injection/recirculation to a hot/cold leg (CCPs; SIPs; RHR), accumulator injection, normal Residual Heat Removal (RHR), letdown, pressurizer spray, or normal charging. |

| Table 3.3-1 (Continued)<br>Summary of Postulated Consequences by System for<br>VEGP-1 and VEGP-2 Piping | |
|---|---|
| RHR - Residual Heat Removal | Mitigating system impacts include loss of RHR functions, sump inventory, and RWST inventory. There are multiple indirect effect scenarios due to jet impingement, spray, pipe whip, and flooding. The interactions are with various cable trays, Containment Spray system, CV and the SI system. |
| SI - Safety Injection | Several segment failures lead to a reactor trip. Mitigating system impacts include loss of SI pump, CCP, accumulator, RHR, sump inventory, RWST, and containment isolation functions. |

**Table 3.4-1**
**Failure Probability Estimates (without ISI)**
**VEGP-1 and VEGP-2 Piping**

| System Description | 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)* | |
| AFW | • Erosion/Corrosion and Thermal Fatigue<br>• Thermal Fatigue<br>• Thermal and Vibrational Fatigue | 7.92E-09 – 2.22E-06<br>3.11E-09 – 6.02E-05<br>4.31E-09 – 4.31E-09 | 1.24E-10 – 3.32E-08<br>2.79E-11 – 5.05E-06<br>2.79E-11 – 1.06E-09 | VEGP has separate AFW nozzles for each Steam Generator. Partial flow is maintained through each nozzle to prevent stratification. AFW piping is in FAC program, which minimizes failure probabilities due to erosion/corrosion. |
| CI | • Thermal Fatigue<br>• Thermal Fatigue and Erosion/Corrosion | 1.25E-08 – 1.69E-05<br>1.11E-02 - 3.33E-02 | SYS - 2.52E-12 – 8.50E-04<br>BREAK - 1.58E-08 - 3.09E-08<br>SYS - 3.28E-06 – 3.33E-02 | Miscellaneous Containment Penetrations – Degradation issues include potential microbiologically induce corrosion in the fire protection line penetration and water hammer in the Nuclear Service Cooling Water lines. |

**Table 3.4-1 (Continued)**
**Failure Probability Estimates (without ISI)**
**VEGP-1 and VEGP-2 Piping**

| System Description | 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)* | |
| CS | • Thermal Fatigue<br>• Thermal and Vibrational Fatigue | 4.25E-07 – 1.21E-05<br>6.47E-07 – 1.03E-05 | 1.63E-10 – 1.86E-06<br>4.47E-10 – 1.45E-06 | This is a standby system at ambient room temperature with RWST water chemistry. Below Stress Corrosion Cracking (SCC) threshold temperature with the VEGP water chemistry. Only cycling and vibration occurs during quarterly pump testing. |
| CV | • Thermal and Vibrational Fatigue and Erosion/Corrosion<br>• Thermal Fatigue<br>• Thermal Fatigue and Vibrational Fatigue | 1.29E-04 – 4.45E-03<br>9.08E-08 – 3.87E-05<br>9.08E-08 – 3.61E-04 | SYS - 6.43E-07 – 2.18E-03<br>SYS - 1.63E-10 – 3.70E-04<br>SLOCA - 1.17E-09 - 1.75E-06<br>MLOCA - 1.14E-09 - 1.19E-06<br>BREAK - 1.07E-09 - 9.34E-07<br>SYS - 1.10E-09 – 1.81E-04 | Highest failure probability is downstream of the flow orifices due to the large pressure drop at the orifices and a postulated potential for wall thinning. |
| FW | • Erosion/Corrosion and Thermal Fatigue<br>• Thermal Fatigue | 4.40E-09 – 1.11E-07<br>1.64E-07 – 2.77E-05 | 1.16E-10 – 1.94E-08<br>4.67E-10 – 2.93E-06 | There are minimal low flow stratification issues for Feedwater because the newer 4-Loop VEGP design has Auxiliary Feedwater nozzles used during low flow conditions. FW piping is in FAC program, which minimizes failure probabilities due to erosion/corrosion. Maximum of 5 thermal fatigue cycles per year on the average based on EPRI FatiguePro™ estimates. |

**Table 3.4-1 (Continued)**
**Failure Probability Estimates (without ISI)**
**VEGP-1 and VEGP-2 Piping**

| System Description | 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)* | |
| MS | • Thermal Fatigue<br>• Thermal and Vibrational Fatigue | 1.33E-08 – 2.45E-07<br>1.48E-06 – 1.87E-05 | 8.73E-10 – 5.71E-05<br>1.60E-08 – 5.06E-06 | Small Leak controlled by vibration of branch lines. Disabling Leak controlled by thermal fatigue in combination with existing linear indications. |
| NS | • Thermal Fatigue | 4.93E-06 – 1.74E-05 | SYS - 1.31E-08 – 9.69E-06<br>SLOCA - 1.59E-08 – 3.33E-07 | Failure probability controlled by thermal fatigue due to cooldown/heatup of lines. |
| RC | • Thermal Fatigue<br>• Thermal Fatigue & Stress Corrosion Cracking<br>• Thermal and Vibrational Fatigue, Stress Corrosion Cracking<br>• Thermal and Vibrational Fatigue | 6.2E-09 – 2.28E-05<br>6.82E-13 – 5.47E-05<br>1.86E-08 - 1.86E-08<br>4.37E-09 - 1.24E-05 | LLOCA - 4.37E-12 – 1.29E-06<br>MLOCA - 7.96E-12 – 1.63E-06<br>SLOCA - 5.10E-11 – 4.48E-06<br>SYS - 3.50E-10 – 2.26E-05<br>LLOCA - 3.14E-11 – 2.17E-05<br>MLOCA - 1.45E-10 – 2.33E-05<br>SLOCA - 1.92E-10 – 1.45E-05<br>SYS - 7.87E-11 – 2.12E-05<br>LLOCA - 3.73E-08 – 7.71E-06<br>MLOCA - 3.73E-08 – 7.71E-06<br>SLOCA - 3.73E-08 – 7.71E-06<br>SYS - 3.73E-08 – 7.71E-06<br>LLOCA - 8.28E-10 – 3.44E-08<br>MLOCA - 1.26E-10 – 2.45E-06<br>SLOCA - 1.39E-10 – 1.46E-05<br>SYS - 2.13E-10 – 1.90E-05 | Based on actual VEGP-2 monitoring data, some thermal stratification occurs in the pressurizer surge line. However, evaluations of the data indicate that thermal stratification has a limited impact on the integrity of the pressurizer surge line.<br>A potential exists for cold water in-leakage and subsequent thermal stratification and striping (NRC Bulletin 88-08) in small lines off the RCS; however, most lines are monitored, which substantially lowers the failure probability from this mechanism.<br>SCC is a potential for the Inconel welds, with the potential evaluated as a function of the temperature. |

**Table 3.4-1 (Continued)**
**Failure Probability Estimates (without ISI)**
**VEGP-1 and VEGP-2 Piping**

| System Description | 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)* | |
| RHR | • Thermal Fatigue<br><br>• Thermal & Vibrational Fatigue | 2.18E-08 – 5.39E-05<br><br>2.30E-06 – 5.39E-05 | SYS - 2.52E-12 – 6.47E-06<br>BREAK - 6.33E-10 - 4.74E-06<br><br>SYS - 2.40E-09 – 5.94E-06<br>BREAK - 3.62E-09 - 5.93E-06 | NRC Bulletin 88-08 Supplement 3 identified potential thermal stratification/striping concerns for RHR piping connected to the RCS. However, monitoring indicates that these concerns are minimal at VEGP. Cracking at the RHR Heat Exchanger Bypass was evaluated and considered in the input. (VEGP-1 was examined with no relevant indications). Failure probability is primarily due to cycling from ambient to 350°F when used for shutdown cooling. (Note: Discharge piping connect to SI, not RCS. |
| SGB | • Erosion/Corrosion and Thermal Fatigue<br><br>• Thermal Fatigue | 3.11E-09 – 5.70E-06<br><br>5.93E-09 – 4.34E-05 | 1.57E-11 – 1.71E-07<br><br>7.16E-11 – 3.39E-06 | Failure probability controlled by thermal fatigue due to cooldown/heatup of line. SGB piping is in FAC program, which minimizes failure probabilities due to erosion/corrosion. |

**Table 3.4-1 (Continued)**
**Failure Probability Estimates (without ISI)**
**VEGP-1 and VEGP-2 Piping**

| System Description | 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)* | |
| SI | • Thermal Fatigue<br>• Thermal Fatigue and Stress Corrosion Cracking<br>• Thermal and Vibrational Fatigue<br>• Thermal and Vibrational Fatigue and Stress Corrosion Cracking | 1.33E-12 – 1.96E-05<br>4.62E-07 – 7.00E-04<br>1.13E-07 – 1.95E-05<br>7.19E-07 - 6.86E-04 | SYS - 5.22E-15 – 3.06E-05<br>BREAK - 5.22E-15 - 3.93E-06<br>SYS - 3.90E-10 – 6.59E-04<br>SYS - 4.19E-12 – 1.63E-06<br>SYS - 8.72E-10 - 2.89E-04<br>BREAK - 5.01E-10 - 5.18E-04 | A large portion of this system is in standby at ambient room temperature with RWST water chemistry. Ambient temperature piping is below SCC threshold temperature with the VEGP water chemistry. Pump discharge is at accumulator pressure due to leaking check valves, but there is no temperature increase or cycling. There is additional SCC potential where SI interfaces with RCS (e.g., Sequoyah cracking) because temperatures are elevated and water is oxygenated. Failure probability controlled by thermal fatigue due to cooldown/heatup of lines interfacing with RHR during shutdown cooling and SCC. |

Notes:
* - Disabling leak rate – Large LOCA (LLOCA), Medium LOCA (MLOCA), Small LOCA (SLOCA), and System Disabling Leak (SYS). When no identifier is shown, this is the System Disabling Leak rate.

| Table 3.5-1<br>Southern Nuclear Operating Company<br>VEGP-1<br>Number of Segments and Piping Risk Contribution by System (without ISI) | | | | | |
|---|---|---|---|---|---|
| System | # of Segments | CDF without Operator Action (/yr) | CDF with Operator Action (/yr) | LERF without Operator Action (/yr) | LERF with Operator Action (/yr) |
| AFW | 48 | 3.92E-11 | 8.73E-11 | 7.25E-14 | 1.64E-13 |
| CI | 169 | 1.22E-09 | 6.11E-11 | 2.06E-12 | 7.94E-13 |
| CS | 66 | 2.51E-10 | 2.32E-10 | 1.01E-12 | 9.47E-13 |
| CV | 274 | 3.52E-08 | 1.91E-10 | 1.46E-10 | 4.20E-12 |
| FW | 34 | 6.72E-12 | 6.72E-12 | 2.60E-12 | 2.60E-12 |
| MS | 80 | 6.76E-09 | 6.76E-09 | 7.47E-11 | 7.47E-11 |
| NS | 9 | 6.85E-11 | 6.85E-11 | 1.36E-13 | 1.36E-13 |
| RCS | 172 | 7.38E-07 | 7.33E-07 | 4.01E-09 | 3.99E-09 |
| RHR | 98 | 4.57E-08 | 6.20E-09 | 4.90E-10 | 5.75E-11 |
| SGB | 80 | 5.67E-11 | 5.67E-11 | 1.06E-13 | 1.06E-13 |
| SI | 150 | 3.07E-09 | 4.98E-10 | 4.82E-11 | 3.94E-11 |
| Total | 1180 | 8.30E-07 | 7.47E-07 | 4.78E-09 | 4.17E-09 |

| Table 3.5-2<br>Southern Nuclear Operating Company<br>VEGP-2<br>Number of Segments and Piping Risk Contribution by System (without ISI) | | | | | |
|---|---|---|---|---|---|
| System | # of Segments | CDF without Operator Action (/yr) | CDF with Operator Action (/yr) | LERF without Operator Action (/yr) | LERF with Operator Action (/yr) |
| AFW | 48 | 3.92E-11 | 8.73E-11 | 7.25E-14 | 1.64E-13 |
| CI | 169 | 1.22E-09 | 6.11E-11 | 2.06E-12 | 7.94E-13 |
| CS | 66 | 2.72E-10 | 2.53E-10 | 1.16E-12 | 1.09E-12 |
| CV | 274 | 3.65E-08 | 2.05E-10 | 1.50E-10 | 5.68E-12 |
| FW | 34 | 6.72E-12 | 1.23E-11 | 2.60E-12 | 2.61E-12 |
| MS | 80 | 6.83E-09 | 6.82E-09 | 7.48E-11 | 7.48E-11 |
| NS | 9 | 6.85E-11 | 6.85E-11 | 1.36E-13 | 1.36E-13 |
| RCS | 172 | 7.38E-07 | 7.33E-07 | 4.02E-09 | 3.99E-09 |
| RHR | 103 | 4.58E-08 | 6.22E-09 | 4.90E-10 | 5.75E-11 |
| SGB | 80 | 5.67E-11 | 5.67E-11 | 1.06E-13 | 1.06E-13 |
| SI | 150 | 3.04E-09 | 5.16E-10 | 4.83E-11 | 3.97E-11 |
| Total | 1185 | 8.32E-07 | 7.48E-07 | 4.79E-09 | 4.17E-09 |

**Table 3.7-1**
**Southern Nuclear Operating Company**
**VEGP-1**
**Summary of Risk Evaluation and Expert Panel Categorization Results**

| System | Number of segments with any RRW >= 1.005 | Number of segments with any 1.001 ≤ RRW < 1.005 | Number of segments with all RRW < 1.001 | Number of segments with any 1.001 ≤ RRW < 1.005 placed in HSS | Number of segments with all RRW < 1.001 selected for inspection | Total number of segments selected for inspection (High Safety Significant Segments) |
|---|---|---|---|---|---|---|
| AFW | 0 | 0 | 48 | 0 | 0 | 0 |
| CI | 0 | 0 | 169 | 0 | 0 | 0 |
| CS | 0 | 3 | 63 | 0 | 3 | 3 |
| CV | 10 | 13 | 251 | 0 | 2 | 2 |
| FW | 0 | 2 | 32 | 0 | 4 | 4 |
| MS | 4 | 12 | 64 | 0 | 14 | 18 |
| NS | 0 | 0 | 9 | 0 | 0 | 0 |
| RCS | 37 | 17 | 118 | 4 | 0 | 41 |
| RHR | 3 | 1 | 94 | 0 | 0 | 3 |
| SGB | 0 | 0 | 80 | 0 | 0 | 0 |
| SI | 2 | 10 | 138 | 1 | 7 | 10 |
| TOTAL | 56 | 58 | 1066 | 5 | 30 | 81 |

**Table 3.7-2**
**Southern Nuclear Operating Company**
**VEGP-2**
**Summary of Risk Evaluation and Expert Panel Categorization Results**

| System | Number of segments with any RRW >= 1.005 | Number of segments with any 1.001 ≤ RRW < 1.005 | Number of segments with all RRW < 1.001 | Number of segments with any 1.001 ≤ RRW < 1.005 placed in HSS | Number of segments with all RRW < 1.001 selected for inspection | Total number of segments selected for inspection (High Safety Significant Segments) |
|---|---|---|---|---|---|---|
| AFW | 0 | 0 | 48 | 0 | 0 | 0 |
| CI | 0 | 0 | 169 | 0 | 0 | 0 |
| CS | 0 | 3 | 63 | 0 | 3 | 3 |
| CV | 10 | 14 | 250 | 0 | 2 | 2 |
| FW | 0 | 2 | 32 | 0 | 4 | 4 |
| MS | 4 | 12 | 64 | 0 | 14 | 18 |
| NS | 0 | 0 | 9 | 0 | 0 | 0 |
| RCS | 37 | 20 | 115 | 4 | 0 | 41 |
| RHR | 3 | 1 | 99 | 0 | 0 | 3 |
| SGB | 0 | 0 | 80 | 0 | 0 | 0 |
| SI | 2 | 11 | 137 | 1 | 7 | 10 |
| TOTAL | 56 | 63 | 1066 | 5 | 30 | 81 |

| TABLE 3.10-1<br>SOUTHERN NUCLEAR OPERATING COMPANY<br>VEGP-1<br>COMPARISON OF CDF/LERF FOR CURRENT SECTION XI AND RISK-INFORMED ISI PROGRAMS AND THE SYSTEMS WHICH CONTRIBUTED SIGNIFICANTLY TO THE CHANGE | | |
|---|---|---|
| Case<br>(Systems Contributing to Change) | Current Section XI | Risk-Informed |
| CDF No Operator Action | 1.37E-07 | 1.34E-07 |
| CV | 2.56E-08 | 2.40E-08 |
| SI | 2.51E-09 | 1.33E-09 |
| CDF with Operator Action | 1.07E-07 | 1.07E-07 |
| RCS | 9.97E-08 | 9.96E-08 |
| RHR | 5.77E-11 | 1.06E-10 |
| SI | 1.29E-10 | 9.71E-11 |
| LERF without Operator Action | 7.51E-10 | 7.38E-10 |
| CV | 1.04E-10 | 9.59E-11 |
| RHR | 5.36E-12 | 9.53E-12 |
| SI | 1.81E-11 | 9.45E-12 |
| LERF with Operator Action | 6.36E-10 | 6.30E-10 |
| SI | 9.53E-12 | 3.27E-12 |

| TABLE 3.10-2<br>SOUTHERN NUCLEAR OPERATING COMPANY<br>VEGP-2<br>COMPARISON OF CDF/LERF FOR CURRENT SECTION XI AND RISK-INFORMED ISI PROGRAMS AND<br>THE SYSTEMS WHICH CONTRIBUTED SIGNIFICANTLY TO THE CHANGE | | |
|---|---|---|
| Case<br>(Systems Contributing to Change) | Current Section XI | Risk-Informed |
| CDF No Operator Action | 1.44E-07 | 1.42E-07 |
| CS | 2.62E-11 | 2.72E-10 |
| CV | 3.28E-08 | 3.09E-08 |
| RCS | 1.01E-07 | 1.00E-07 |
| RHR | 7.84E-10 | 1.08E-09 |
| CDF with Operator Action | 1.08E-07 | 1.08E-07 |
| CS | 6.76E-12 | 2.53E-10 |
| CV | 6.07E-11 | 2.04E-10 |
| RCS | 1.01E-07 | 1.00E-07 |
| RHR | 3.40E-11 | 1.07E-10 |
| SI | 1.21E-10 | 1.02E-10 |
| LERF without Operator Action | 7.79E-10 | 7.69E-10 |
| CV | 1.35E-10 | 1.25E-10 |
| RCS | 5.46E-10 | 5.45E-10 |
| RHR | 4.79E-12 | 9.54E-12 |
| SI | 1.35E-11 | 9.37E-12 |
| LERF with Operator Action | 6.37E-10 | 6.33E-10 |
| CS | 2.57E-13 | 1.09E-12 |
| CV | 5.21E-12 | 5.57E-12 |
| RCS | 5.46E-10 | 5.45E-10 |
| RHR | 2.01E-13 | 6.67E-13 |
| SI | 7.43E-12 | 3.26E-12 |

**TABLE 5-1a**
**VEGP-1**
**STRUCTURAL ELEMENT SELECTION**
**RESULTS AND COMPARISON TO ASME SECTION XI**
**1989 EDITION REQUIREMENTS**

| System | Number of High Safety Significant Segments (No. of HSS in Augmented Program [d] / Total No. of Segments in Augmented Program) | Degradation Mechanism(s) | Class | ASME Code Category | Weld Count [h] | | ASME XI Examination Methods (Volumetric (Vol) and Surface (Sur)) | | RI-ISI [a] | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Butt | Socket | Vol & Sur | Sur Only | SES Matrix Region | Number of Exam Locations |
| 1AFW[e] | 0 | E/C, MF, TF<br>MF, TF<br>MF, TF, VF | Class 2 | C-F-2 | 178 | 0 | 14 | 0 | - | 0 |
| 1CI | 0 | MF, TF<br>MF, TF, E/C | Class 2 | (b) | (b) | (b) | (b) | (b) | - | 0 |
| 1CS | 3 | MF, TF<br>MF, TF, VF | Class 2 | C-F-1 | 216 | 0 | 19 | 0 | 2 | 3 + 1 [g] |
| 1CV | 2 | E/C, MF, TF, VF | Class 1 | B-J | 68 | 21 | 0 | 37 | - | 0 |
| | | MF, TF | Class 2 | C-F-1 | 307 | 3 | 29 | 4 | 2 | 1 + 5 [g] |
| | | MF, TF, VF | Class 2 | (c) | (c) | (c) | (c) | (c) | 2 | 1 |
| 1FW[e] | 4(4/16) | MF, TF, E/C<br>MF, TF | Class 2 | C-F-2 | 114 | 0 | 7 | 2 | 1A | 8 |

**TABLE 5-1a (Continued)**
**VEGP-1**
**STRUCTURAL ELEMENT SELECTION**
**RESULTS AND COMPARISON TO ASME SECTION XI**
**1989 EDITION REQUIREMENTS**

| System | Number of High Safety Significant Segments (No. of HSS in Augmented Program [d] / Total No. of Segments in Augmented Program) | Degradation Mechanism(s) | Class | ASME Code Category | Weld Count [h] | | ASME XI Examination Methods (Volumetric (Vol) and Surface (Sur)) | | RI-ISI [a] | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Butt | Socket | Vol & Sur | Sur Only | SES Matrix Region | Number of Exam Locations |
| 1MS | 18(4/12) | MF, TF<br>MF, TF, VF | Class 2 | C-F-2 | 157 | 0 | 15 | 0 | 1A, 2 | 24 |
| | | | Class 2 | (c) | (c) | (c) | (c) | (c) | 2 | 6 |
| 1NS | 0 | MF, TF | Class 2 | (b) | (b) | (b) | (b) | (b) | - | 0 |
| 1RCS [f] | 41 | MF, TF<br>MF, TF, SCC<br>MF, TF, VF, SCC<br>MF, TF, VF | Class 1 | B-F | 14 | 0 | 14 | 0 | 2 | 14 |
| | | | Class 1 | B-J | 294 | 28 | 67 | 44 | 2 | 30 |
| | | | Class 1 | (c) | (c) | (c) | (c) | (c) | - | 2[g] |
| 1RHR | 3 | MF, TF<br>MF, TF, VF | Class 2 | C-F-1 | 407 | 0 | 34 | 0 | 2 | 3 |
| 1SGB[e] | 0 | E/C. MF, TF<br>MF, TF | Class 2 | (b) | (b) | (b) | (b) | (b) | - | 0 |

**TABLE 5-1a (Continued)**
**VEGP-1**
**STRUCTURAL ELEMENT SELECTION**
**RESULTS AND COMPARISON TO ASME SECTION XI**
**1989 EDITION REQUIREMENTS**

| System | Number of High Safety Significant Segments (No. of HSS in Augmented Program [d] / Total No. of Segments in Augmented Program) | Degradation Mechanism(s) | Class | ASME Code Category | Weld Count [h] | | ASME XI Examination Methods (Volumetric (Vol) and Surface (Sur)) | | RI-ISI [a] | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Butt | Socket | Vol & Sur | Sur Only | SES Matrix Region | Number of Exam Locations |
| 1SI | 10 | MF, TF<br>MF, TF, SCC<br>MF, TF, VF<br>MF, TF, VF, SCC | Class 1 | B-J | 490 | 24 | 46 | 64 | 2 | 8 + 5 [g] |
| | | | Class 2 | C-F-1 | 561 | 0 | 42 | 1 | 2 | 2 |
| TOTAL | 81(8/28) | | Class 1 | B-F | 14 | 0 | 14 | 0 | 2 | 14 NDE |
| | | | | B-J | 852 | 73 | 113 | 145 | 2 | 43 NDE |
| | | | | (c) | (c) | (c) | (c) | (c) | - | 2 VISUAL |
| | | | Class 2 | C-F-1 | 1491 | 3 | 124 | 5 | 2 | 12 NDE+3 VISUAL |
| | | | | C-F-2 | 449 | 0 | 36 | 2 | 1A, 2 | 32 NDE |
| | | | | (b) | (b) | (b) | (b) | (b) | - | 0 |
| | | | | (c) | (c) | (c) | (c) | (c) | 2 | 7 NDE |
| | | | Total | | 2806 | 76 | 287 | 152 | | 108 NDE + 5 VISUAL |

**TABLE 5-1b**
**VEGP-2**
**STRUCTURAL ELEMENT SELECTION**
**RESULTS AND COMPARISON TO ASME SECTION XI**
**1989 EDITION REQUIREMENTS**

| System | Number of High Safety Significant Segments (No. of HSS in Augmented Program [d] / Total No. of Segments in Augmented Program) | Degradation Mechanism(s) | Class | ASME Code Category | Weld Count [h] | | ASME XI Examination Methods (Volumetric (Vol) and Surface (Sur)) | | RI-ISI [a] | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Butt | Socket | Vol & Sur | Sur Only | SES Matrix Region | Number of Exam Locations |
| 2AFW [e] | 0 | E/C, MF, TF<br>MF, TF<br>MF, TF, VF | Class 2 | C-F-2 | 182 | 0 | 14 | 0 | - | 0 |
| 2CI | 0 | MF, TF<br>MF, TF, E/C | Class 2 | (b) | (b) | (b) | (b) | (b) | - | 0 |
| 2CS | 3 | MF, TF<br>MF, TF, VF | Class 2 | C-F-1 | 204 | 0 | 17 | 0 | 2 | 3 |
| 2CV | 2 | E/C, MF, TF, VF | Class 1 | B-J | 79 | 23 | 0 | 39 | - | 0 |
| | | MF, TF | Class 2 | C-F-1 | 285 | 1 | 17 | 0 | 2 | 1 + 2 [g] |
| | | MF, TF, VF | Class 2 | (c) | (c) | (c) | (c) | (c) | 2 | 1 |
| 2FW [e] | 4(4/16) | MF, TF, E/C<br>MF, TF | Class 2 | C-F-2 | 111 | 0 | 9 | 1 | 1A | 8 |

**TABLE 5-1b (Continued)**
**VEGP-2**
**STRUCTURAL ELEMENT SELECTION**
**RESULTS AND COMPARISON TO ASME SECTION XI**
**1989 EDITION REQUIREMENTS**

| System | Number of High Safety Significant Segments (No. of HSS in Augmented Program [d] / Total No. of Segments in Augmented Program) | Degradation Mechanism(s) | Class | ASME Code Category | Weld Count [h] | | ASME XI Examination Methods (Volumetric (Vol) and Surface (Sur)) | | RI-ISI [a] | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Butt | Socket | Vol & Sur | Sur Only | SES Matrix Region | Number of Exam Locations |
| 2MS | 18(4/12) | MF, TF | Class 2 | C-F-2 | 159 | 0 | 15 | 0 | 1A, 2 | 25 |
| | | MF, TF, VF | Class 2 | (c) | (c) | (c) | (c) | (c) | 2 | 6 |
| 2NS | 0 | MF, TF | Class 2 | (b) | (b) | (b) | (b) | (b) | - | 0 |
| 2RCS [f] | 41 | MF, TF | Class 1 | B-F | 14 | 0 | 14 | 0 | 2 | 14 |
| | | MF, TF, SCC | Class 1 | B-J | 304 | 28 | 77 | 43 | 2 | 30 |
| | | MF, TF, VF, SCC | Class 2 | (c) | (c) | (c) | (c) | (c) | - | 2 [g] |
| | | MF, TF, VF | | | | | | | | |
| 2RHR | 3 | MF, TF | Class 2 | C-F-1 | 405 | 0 | 31 | 0 | 2 | 3 |
| | | MF, TF, VF | | | | | | | | |
| 2SGB [e] | 0 | E/C. MF, TF | Class 2 | (b) | (b) | (b) | (b) | (b) | - | 0 |
| | | MF, TF | | | | | | | | |

**TABLE 5-1b (Continued)**
**VEGP-2**
**STRUCTURAL ELEMENT SELECTION**
**RESULTS AND COMPARISON TO ASME SECTION XI**
**1989 EDITION REQUIREMENTS**

| System | Number of High Safety Significant Segments (No. of HSS in Augmented Program [d] / Total No. of Segments in Augmented Program) | Degradation Mechanism(s) | Class | ASME Code Category | Weld Count [h] | | ASME XI Examination Methods (Volumetric (Vol) and Surface (Sur)) | | RI-ISI [a] | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Butt | Socket | Vol & Sur | Sur Only | SES Matrix Region | Number of Exam Locations |
| 2SI | 10 | MF, TF<br>MF, TF, SCC<br>MF, TF, VF<br>MF, TF, VF, SCC | Class 1 | B-J | 490 | 24 | 41 | 65 | 2 | 8 + 5 [g] |
| | | | Class 2 | C-F-1 | 530 | 0 | 41 | 0 | 2 | 2 |
| TOTAL | 81 (8 / 28) | | Class 1 | B-F | 14 | 0 | 14 | 0 | 2 | 14 NDE |
| | | | | B-J | 873 | 75 | 118 | 147 | 2 | 43 NDE |
| | | | Class 2 | C-F-1 | 1424 | 1 | 106 | 0 | 2 | 9 NDE + 2 VISUAL |
| | | | | C-F-2 | 452 | 0 | 38 | 0 | 1A, 2 | 33 NDE |
| | | | Class 2 | (b) | (b) | (b) | (b) | (b) | - | 0 |
| | | | Class 2 | (c) | (c) | (c) | (c) | (c) | 2 | 7 NDE + 2 VISUAL |
| | | | Total | | 2763 | 76 | 276 | 147 | | 106 NDE + 4 VISUAL |

**TABLE 5-1b (Continued)**
**VEGP-2**
**STRUCTURAL ELEMENT SELECTION**
**RESULTS AND COMPARISON TO ASME SECTION XI**
**1989 EDITION REQUIREMENTS**

Summary: Current ASME Section XI selects a total of 439 non-destructive exams for VEGP-1 while the RI-ISI program selects a total of 113 exams (5 visual exams), which results in a 74% reduction.

Current ASME Section XI selects a total of 423 non-destructive exams for VEGP-2 while the RI-ISI program selects a total of 110 exams (4 visual exams), which results in a 74% reduction

Degradation Mechanisms: VF – Vibratory Fatigue; TF – Thermal Fatigue; MF – Mechanical Fatigue; E/C – Erosion/Corrosion; SCC – Stress Corrosion Cracking

Notes for Table 5-1a and 5-1b.

a. System pressure test requirements and VT-2 visual examinations shall continue to be performed in ASME Code Class 1 and 2 systems as described in the current ASME Section XI program.
b. Piping is exempt per the requirements of the 1989 Edition of Section XI and there are no RI-ISI examinations.
c. Piping is exempt per the requirements of the 1989 Edition of Section XI; however, examinations are required per the proposed RI-ISI program.
d. Augmented program consists of those high-energy welds examined in Feedwater and Main Steam ("No-Break" Zone). Examinations will continue per Technical Specification requirements.
e. Thickness measurements continue to be performed in the AFW, FW, and SGB systems as part of the flow-accelerated corrosion (FAC) program (also known as erosion/corrosion).
f. Monitoring program continues for high-cycle thermal fatigue (striping/stratification) issues.
g. 13 piping segments for VEGP-1 were added for change in risk considerations: One CS NDE exam, two CV NDE exams, three CV VT exams, two RC VT exams, and five SI NDE exams.
   9 piping segments for VEGP-2 were added for change in risk considerations: Two CV VT exams, two RC VT exams, and five SI NDE exams.
h. Estimated weld counts based on the VEGP ISI database.