Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

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

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Request to Implement a Risk-Informed Inservice
Testing Program for Pumps and Valves
Beginning the Second 10-Year Interval (Relief Request RR-ENG-IST-2-01)

In accordance with the provisions of 10CFR50.55a(a)(3)(i), the South Texas Project requests Nuclear Regulatory Commission approval to use an alternative approach to the ASME Code requirements for determining the testing intervals for pumps and valves. Attachment 1, "South Texas Project Risk-Informed Inservice Testing Program for Pumps and Valves" is a complete description and analysis of the proposed method and contains the supporting bases for this alternative to the ASME Section XI Code for determining test intervals. The Risk-Informed Inservice Testing Program defined in this submittal follows the criteria of Regulatory Guide 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing." The alternate method will provide an acceptable level of quality and safety as required by Regulatory Guide 1.175 because key safety principles of defense-in-depth and safety margins are maintained.

The South Texas Project will begin the second 10-year inservice testing interval no later than December 1, 2001. The South Texas Project has updated the Inservice Testing Program and is now testing pumps and valves in accordance with the 1989 Edition of the Section XI Code, which invokes by reference the 1987 Edition of the O&M Code with 1988 Addenda. During the second 10-year interval, the South Texas Project will continue to comply with the 1989 Edition of the ASME Section XI Code for pumps and valves, except the test intervals will be determined by the Risk-Informed Inservice Testing Program described in this submittal.

The engineering analysis described in Attachment 1 provides the basis for the South Texas Project Risk-Informed Inservice Testing Program for Pumps and Valves. The following table indicates how various sections of Attachment 1 address Regulatory Guide 1.175.

A047

Section of Attachment 1	Subject
1.1	A description of the changes associated with the proposed RI-IST Program
2.1.1	Identification of any changes to the plant's design, operations, and other activities associated with the proposed RI-IST program and the basis for the acceptability of these changes
2.3.2	The process used to identify candidates for reduced and enhanced IST requirements, including a description of the categorization of components using the PRA and the associated sensitivity studies
2.3.1, 2.3.2	A description of the PRA used for the categorization process and for the determination of risk impact, in terms of the process to ensure quality and the scope of the PRA, and how compensation is provided in the integrated decision-making process for limitations in quality, scope, and level of detail
2.3.3	A description of how the impact of the change is modeled in the IST components (including a quantitative or qualitative treatment of component degradation) and a description of the impact of the change on plant risk in terms of CDF and LERF and how this impact compares with the decision guidelines
2.2	A discussion of how the key principles were (and will continue to be) maintained
2.4	The integrated decision-making process used to help define the RI-IST program, including any decision criteria used
2.1.2	A summary of previously approved relief requests for components categorized as HSSC along with exemption requests, technical specification changes, and relief requests needed to implement the proposed RI-IST Program
2.1.2	An assessment of the appropriateness of previously approved relief requests

Attachment 2 to this letter is the "Risk-Informed Inservice Testing Program Description Summary." This attachment describes the requirements for the categorization of components using the Probabilistic Risk Assessment inputs and the blending of deterministic information in an Integrated Decisionmaking Process. Additionally, Attachment 2 describes the development of test frequencies and testing methodologies and describes the evaluation of cumulative risk impact of testing changes. The implementation, monitoring and corrective action plans, period assessments of the program, and a method for making changes to the program are also described in this attachment.

Attachment 3 contains four reports from the Risk-Informed Inservice Testing database. These reports include valve and pump lists that provide the scope of the inservice testing plan for the second 10-year interval.

If there are any questions, please contact either M. S. Lashley at (361) 972-7523 or me at (361) 972-7902.

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BJS/PLW

Attachments:
1. Risk-Informed Inservice Testing Program Engineering Analysis
2. Risk-Informed Inservice Testing Program Description Summary
3. Valve and Pump Lists for 2nd 10-Year Interval

cc:

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U. S. Nuclear Regulatory Commission
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Attachment 1

RISK-INFORMED INSERVICE TESTING PROGRAM

FOR

PUMPS AND VALVES

ENGINEERING ANALYSIS

Inservice Testing Program Coordinator	5/9/01 *Date*
Supervisor, Testing/Programs Engineering	5/10/01 *Date*

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ACRONYMS

ACRONYM	DESCRIPTION
AF	Auxiliary Feedwater System
AOV	Air or Pneumatic Valve
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel Code
BAT	Boric Acid Transfer [pump]
CAP	Corrective Action Program
CCF	Common Cause Failure
CCW	Component Cooling Water System
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CPSES	Comanche Peak Steam Electric Station (TU Electric)
CR	Condition Report
CV	Check Valve
CVP	Check Valve Program
ECW	Essential Cooling Water System
EP	Expert Panel
FV	Fussell-Vesely
GQA	Graded Quality Assurance
HSSC	High Safety Significant Component– High Fussell-Vesely
IDP	Integrated Decisionmaking Process
IPE	Individual Plant Examination
IPEEE	Individual Plant External Events Examination
IST	Inservice Testing
JOG	Joint Owners Group
LERF	Large Early Release Frequency
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident
LSSC	Lower Safety Significant Component- Low Fussell-Vesely and Low Risk Achievement Worth
MGL	Multiple Greek Letter
MOV	Motor-Operated Valve
MS	Main Steam System
NPRDS	Nuclear Plant Reliability Data System
NRC	Nuclear Regulatory Commission
OEG	Operating Experience Group
PORV	Power-Operated Relief Valve
PRA	Probabilistic Risk Assessment
RAW	Risk Achievement Worth
RHR	Residual Heat Removal System
RI-IST	Risk-Informed Inservice Testing

ACRONYM	DESCRIPTION
RV	Relief Valve
SBO	Station Blackout
SER	Safety Evaluation Report
SI	Safety Injection System
SONGS	San Onofre Nuclear Generating Station (Southern California Edison)
SGTR	Steam Generator Tube Rupture
STP	South Texas Project
SSC	Structure, System, or Component
TS	Technical Specifications
TXU	Texas Utilities
UFSAR	Updated Final Safety Analysis Report
WG	Working Group
WOG	Westinghouse Owners Group

EXECUTIVE SUMMARY

The South Texas Project (STP) submits this report to the U.S. Nuclear Regulatory Commission (NRC) for approval of a risk-informed Inservice Testing (RI-IST) program for pumps and valves at STP Units 1 and 2. The program outline conforms to the NRC-approved methods and Regulatory Guides[1,2]. The methodology employed in the development of this program bears close resemblance to that implemented by the NRC-approved RI-IST pilot program at Texas Utilities' (TXU) Comanche Peak Steam Electric Station (CPSES) and the NRC-approved program at Southern California Edison's San Onofre Nuclear Generating Station (SONGS). Furthermore, this program incorporates insights from the Safety Evaluation Reports (SERs) for both programs[3,4].

Given the reliance on insights derived from the Probabilistic Risk Assessment (PRA), the risk assessment satisfies industry standards associated with PRA. The PRA has been used in support of other risk-informed applications at STP and has been deemed to be of a quality consistent with that required to perform accurate, thorough, and comprehensive evaluations for a RI-IST application. The inclusion of inservice testing (IST) program effects on cumulative plant risk is comprehensive. This quantitative evaluation of key RI-IST program elements includes the effects of compensatory measures, the influence of staggered testing on common cause failure (CCF), and the beneficial effect of enhanced IST testing strategies on risk.

A key element of the RI-IST program is the Integrated Decisionmaking Process (IDP). STP's IDP is comprehensive, ensuring that key safety principles such as defense-in-depth and safety margins are maintained. The process considered relevant component-specific information, including design basis safety functions, PRA risk importance, and a detailed analysis of component corrective maintenance history. Therefore, the Integrated Decisionmaking Process assures a detailed evaluation and Panel approval of component categorization results and supporting studies.

Further, insights from the Integrated Decisionmaking Process support the conclusion that several safety enhancements to a plant IST program can be derived, both directly and indirectly, by implementing the results of the probabilistic and deterministic approach presented in this report. These safety benefits have been treated both quantitatively and qualitatively, providing a reasonable and justifiable basis for implementing the program discussed herein.

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1 Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant-specific Changes to the Licensing Basis," July 1998.

2 Regulatory Guide 1.175, "An Approach for Plant-specific, Risk-informed Decisionmaking: Inservice Testing," August 1998.

3 "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to the TU Electric Request to Implement a Risk-informed Inservice Testing Program at Comanche Peak Steam Electric Station (CPSES), Units 1 And 2, Docket Numbers 50-445 And 50-446."

4 "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to the Southern California Edison Request to Implement a Risk-informed Inservice Testing Program at San Onofre Nuclear Generating Station, Units 2 and 3, Docket Numbers 50-361 and 50-362."

Background

The intent of current IST programs is to include all active, safety-related pumps and valves that are credited in the plant design basis safety analysis. In general, the IST equipment lists are developed by review of plant drawings showing ASME Code Class 1, 2, and 3 classification boundaries. All components within the boundaries are then reviewed to determine whether or not they have been credited with an active safety function under the plant licensing basis. The Updated Final Safety Analysis Report (UFSAR) analyses and other design basis documentation provide the primary bases for these determinations.

After publication of its policy statement[5] on the use of probabilistic risk assessment (PRA) in nuclear regulatory activities, the Commission directed the NRC staff to develop regulatory guidance that incorporates risk insights. Concurrently, industry risk-informed pilot projects explored the process for supplementing traditional engineering approaches in reactor regulation with probabilistic information. This effort has culminated in several relevant and extremely significant regulatory advances in the area of risk-informed applications:

1. Issuance of Regulatory Guide (RG) 1.174[1] and companion regulatory guidance (including RG 1.175[2]), which provide the regulatory framework to fashion an inservice testing program that focuses resources on risk-significant pumps and valves,
2. NRC acceptance of TXU's CPSES relief request[3], one of the industry risk-informed IST pilot projects,
3. NRC acceptance of SCE's SONGS relief request[4], one of the follow-on risk-informed IST projects,
4. NRC acceptance of STP's graded quality assurance (GQA) program[6], and
5. NRC draft acceptance of some aspects of STP's request for exemptions from special treatment requirements[7].

As has been demonstrated during the CPSES and SONGS RI-IST projects, improvements to IST programs using a risk-informed approach can reduce operating costs while maintaining a high level of plant safety. Possible benefits from improved IST programs include reduced costs associated with inservice testing, as well as:

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[5] Nuclear Regulatory Commission, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," Federal Register, Vol. 60, No. 158, August 16, 1995.

[6] "Safety Evaluation by the Office of Nuclear Reactor Regulation [Related to the] Houston Lighting and Power Company South Texas Project, Units 1 and 2, Graded Quality Assurance Program, Docket Numbers 50-498 and 50-499."

[7] "Safety Evaluation by the Office of Nuclear Reactor Regulation, Risk-informed Exemptions from Special Treatment Requirements, STP Nuclear Operating Company, South Texas Project Electric Generation Station, Units 1 and 2, Docket Nos. 50-498 and 50-499."

- Less time required to perform the tests and analyze results;
- Reduced costs of specialized test equipment or vendor services;
- Fewer possible effects on critical path outage duration; and
- Less radiation exposure.

For these reasons it is advantageous for utilities to pursue IST program improvements. The impact of changes on plant safety is of primary interest and is the controlling factor in implementing such changes. However, changes that negligibly affect plant safety should not be ruled out, especially if such changes can lead to significant plant performance improvements in other areas.

Project Scope

The scope of this project is to build a RI-IST program for STP Units 1 and 2, one which optimizes safety benefits in ensuring pump and valve performance. The project applies a risk-informed approach for performing a comprehensive IST program review and for proposing program enhancements. The principal results of the project are recommendations for adjustments to test frequency intervals for a large percentage of IST components. The project focuses on optimizing the overall component test schedule by applying resources commensurate with the component safety function, performance, and relative risk. In this study, all components within the scope of the IST program were examined. However, only those determined to be less safety significant have been considered for Code relief. The more safety significant components have been reviewed by component experts to ensure that the appropriate tests have been identified and are performed on those components for their respective failure modes.

Project Approach

The STP risk-informed IST project was developed and implemented by Nuclear Engineering's Testing/Programs Engineering Division with PRA support provided by the Risk and Reliability Analysis Group. A multi-discipline RI-IST Working Group served as integrated decision-makers, assessing information provided by the project team (i.e., risk measures and component performance history), and considering component categorization information produced by other plant risk-informed programs to arrive at an overall RI-IST rank and supporting narrative basis for each component group analyzed. In addition, a cross-functional plant Expert Panel, as well as industry experts who participated in both the TXU and SCE risk-informed IST projects, worked to facilitate and guide the process to ensure a consistent and scrutable outcome. The STP project employed a method that blended probabilistic and traditional engineering insights to identify opportunities to reduce those IST-related regulatory requirements and commitments that require significant resources to comply with and/or implement, but contribute insignificantly to safe and reliable operation. Using risk-informed technologies, the project determined the safety significance of IST components, as well as components not in the IST program. A combination of deterministic and risk-informed methods was applied to determine testing intervals and compensatory measures that correspond to each component's safety significance. The results of the project provide the basis for this request to the NRC to approve implementation of an alternate testing strategy.

Overall project objectives and milestones were established by key risk-informed IST project members. The project was divided into the five major tasks listed below:

- Component Function Evaluation
- Component Corrective Maintenance Evaluation
- Calculation of Risk Measures Using the STP PRA
- Component Risk Categorization by Working Group and Review by Expert Panel
- Cumulative Risk Evaluation Using the STP PRA

The component function evaluation established the design basis safety functions of IST components and related these functions to component failure modes modeled by the PRA. Modeling implications were also identified, including the component or system-level assumptions that affect the level of credit the PRA affords an IST component's safety function. The component corrective maintenance evaluation validated the basis for the PRA reliability assessment and demonstrated how it compared to generic and plant-specific experience. It also established a baseline for future monitoring that is needed to compensate for some of the components whose testing frequency requirements are reduced.

The PRA was then used in a variety of ways to evaluate the safety significance of components and their functions. Sensitivity studies demonstrated the robustness of the methods and the results. This process was followed by the RI-IST Working Group review and validation of the PRA risk measure, a process that ensured an integrated effort through active technology transfer. The Working Group consisted of members with expertise in the areas of power plant operations, plant maintenance, PRA, nuclear safety analysis, systems engineering, design basis engineering, quality assurance, licensing, and Inservice Testing (including ASME B&PV Code Section XI and ASME Code Cases). In addition to considering the basis for the PRA risk measure for modeled components, the Working Group qualitatively assessed the following for each component group:

- The degree to which component failure leads to an increase in the frequency of initiating events,
- The degree to which component failure leads to the failure of another safety system,
- The degree to which component failure causes a transient,
- The role of the component in the plant Emergency Operating Procedures (EOPs), and
- The role of the component in plant shutdown.

As part of the process, the Working Group authored a narrative basis to support the final RI-IST categorization of each component group.

Subsequent to the Working Group initial RI-IST categorization of components, the STP plant Expert Panel considered and ultimately validated the results of all Working Group activities and studies performed by the IST project members. The Expert Panel consisted of members with expertise in the areas of power plant operations, plant maintenance, PRA, nuclear safety analysis, design basis engineering, and quality assurance. The Expert Panel served as the central point of decision-making for major technical issues and offered guidance to risk-informed IST project members in performing their work.

It was concluded that the strength of this risk-informed IST program and the integrity of its results lie both in the robustness of the methodology and in the quality and work of the RI-IST Working Group and plant Expert Panel. This integrated decision-making process was implemented according to clear guidelines and operated directly from documentation produced in earlier tasks.

All project tasks were conducted with reproducibility and retrievability in mind. The project deliverables – including tables of IST functions, PRA functions, PRA risk measures, component ranking outcomes, component functional failures, RI-IST Working Group decision bases, valve groups, test interval information, and monitoring requirements--are housed in a database from which the IST engineer may administer the risk-informed IST program.

Conformance with Key Safety Principles

The proposed RI-IST program meets all acceptance criteria and guidance specified in RG 1.174 and RG 1.175, including the four element approach to evaluating proposed changes in Section 2 of RG 1.174. These acceptance criteria include the five principles of integrated decision-making discussed in Figure 1 of RG 1.174, such as maintaining defense-in-depth and safety margins. In addition, several safety benefits to the plant IST program can be derived both directly and indirectly.

Direct Safety Enhancements

Possibly the most important safety benefit resulting from application of the RI-IST methodology at STP is the promotion of an environment in which participants are encouraged to evaluate current testing strategies and, in particular, the effectiveness of those strategies to detect potential challenges to safety. If another testing strategy exists for a highly safety significant or medium safety significant component, participants feel obliged to consider whether this strategy provides an enhanced understanding of the component's ability to perform its safety function during a design basis accident scenario. For example, a revised testing strategy for the Low Head Safety Injection (LHSI) pumps will be an important safety effect due to the potential core damage frequency (CDF) improvement value of these components. Currently, these components are tested in a mini-flow configuration, which can be potentially damaging to components on the line over a sustained period of time (i.e., with regard to vibration tests). STP proposes to replace the quarterly mini-flow test with a test performed at full flow conditions during refueling outages. This test is generally considered to be much more effective at detecting degradation that could potentially lead to failure of the component to perform its safety function than the current test. Furthermore, as the full flow test requires that components perform their functions at design or near design conditions (i.e., the optimum testing environment), this test is generally considered by industry experts to be less damaging to active components. If inclusion of the full flow test leads to better knowledge of the capability of the pump, one could conservatively postulate an improvement in the CDF resulting from this enhanced test strategy.

In general, relaxing IST intervals for many lower priority components allows STP to focus greater attention and resources on high priority IST components. A resource reallocation of this nature could translate into many direct safety enhancements. Test requirements associated with the high priority group of IST components are expected to be more rigorous and demanding in nature than for the other groups. These requirements provide added assurance that any problems that may impact the functionality of the components will be identified and resolved expeditiously. Second, the resulting risk-informed IST program will consider whether some risk-significant components that are outside the scope of ASME Code Classes 1, 2, and 3 should be added to the IST program to improve safety. Finally, because extensive testing can have adverse safety and operational consequences, reduction of testing may reduce component wear-out and operator burden. These changes are expected to improve safety.

Indirect Safety Enhancements

There are other indirect safety benefits to this approach that are as important. Risk-informed prioritization efforts identify the safety-significant IST components and the impact of their potential failures on plant safety. In addition, these analyses identify important scenarios that provide information with respect to the operational demand that may be placed on a given component. Such information is valuable because it relates the performance of the IST component to the broader context of plant safety. This allows more rational decision-making, more efficient use of resources, and is central to optimizing safety benefits.

RI-IST Project Results

Component categorization of Unit 1 IST valves and pumps yielded the following results:

RISK RANKING	PERCENTAGE OF COMPONENTS[8] (UNIT 1)
RI-IST High	10.3% (56 components)
RI-IST Medium	15.5% (84 components)
RI-IST Low	69.2% (375 components)

According to the above table, 84.7% of the ranked components are eligible for interval extension. Although the engineering analysis was performed for components in both Units 1 and 2, the tabular reports in Attachment 3 (e.g., "Valves in the IST Program" and "RI-IST Component Categorizations and Test Frequencies") list only Unit 1 components. Unit 1 component functions mirror Unit 2 component functions, so the tables reflect information that applies to components in both units. When the performance history of a component group on one unit dictated a more conservative extension, that

[8] Containment isolation valves to be tested per 10 CFR 50, Appendix J, Option B account for less than 5% (27 components) of the Unit 1 IST components.

extension was applied to both units.

Upon implementation of the program, safety enhancements are expected from focusing resources on RI-IST High components and reducing the testing frequency on RI-IST Medium and RI-IST Low components, as discussed above. Because extensive testing on RI-IST Medium and RI-IST Low components may adversely impact safety, reduction of testing should reduce component wear-out, operator burden, system unavailability, cost of testing, and radiation exposure. Reduced testing could also achieve an optimum balance between the positive impacts of testing and the negative effects of removing equipment from service and entering a less than optimum plant configuration, that have the potential to result in valve misalignments. Focusing of resources on RI-IST High and Medium components includes improved testing of LHSI pumps and enhanced testing of selected components, such as motor operated valves (MOVs) (diagnostic testing) and pumps (including performance monitoring activities, such as spectral analysis and thermography), beyond Code testing requirements. The cumulative effects from reduced testing of RI-IST Low and RI-IST Medium components and enhanced testing of selected RI-IST High components are tangible risk benefits which were not used in quantifying the risk impact of the risk-informed IST program.

Given the relaxation of test intervals, the addition of components to the program and the non-quantified tangible risk benefits, the impact of the proposed RI-IST program will be risk neutral.

1.0 PROPOSED CHANGES

1.1 DESCRIPTION OF PROPOSED CHANGES

STP Technical Specification (TS) 4.0.5 requires that inservice testing of ASME Code Class 1, 2, and 3 pumps and valves be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (ASME Code) and applicable Addenda as required by 10CFR50.55a(f). Additionally, 10CFR50.55a(f)(4)(ii) requires that the Inservice Testing program be updated during successive 120-month intervals to comply with the new code of record incorporated by reference in paragraph (b) of the regulation. As previously submitted and approved[9], the South Texas Project has updated the Inservice Testing Program and is now testing pumps and valves in accordance with the 1989 Edition of the Section XI Code, which references the 1987 Edition and 1988 Addenda of the O&M Code. The South Texas Project will continue testing in accordance with the 1989 Section XI Code for pumps and valves. This submittal requests approval to implement an alternative method for the determination of test intervals. This alternative method is consistent with acceptance criteria and guidance contained in Regulatory Guides 1.174 and 1.175, and provides an acceptable level of quality and safety in accordance with 10CFR50.55a(a)(3)(i).

STP's proposed RI-IST program addresses the majority of the 1376 pumps and valves in the current Code-required IST program, including MOVs, check valves (CVs), air-operated valves (AOVs), manual valves and the Main Steam Safety Valves and Reactor Coolant system Pressurizer Safety Valves. STP has updated the IST program to include the testing of relief valves pursuant to the 1989 Section XI Code. Specifically, 90 relief valves in each unit have been added to the program and will be tested in accordance with ASME/ANSI OM-1987 Part 1 with the associated 10-year staggered testing interval commitment. The new relief valves and skid-mounted valves were excluded from the risk-ranking process because STP plans to continue to test these components at current Code-prescribed test intervals. The skid-mounted valves are tested in accordance with ASME/ANSI OM 1987, OMa 1988 Addenda, Part 10, in concert with the guidance presented in NUREG-1482 relative to skid-mounted components. For example, the Diesel Generator skid-mounted valves are tested monthly according to current diesel generator testing protocol.

In lieu of performing inservice tests on pumps and valves whose function is required for safety at frequencies specified in the ASME Code, as required by 10CFR50.55a(f)(4)(ii) for the second 120-month interval, STP presents an alternative testing strategy. The alternative would allow the inservice test strategies of those pumps and valves to be determined in accordance with the following guidelines, which are consistent with the guidelines established in recently approved RI-IST programs at Texas Utilities' Comanche Peak Steam Electric Station and Southern California Edison's San Onofre Nuclear Generating Station:

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[9] NRC Correspondence dated March 15, 1999, Inservice Testing Program Relief Request RR-17, South Texas Project, Units 1 and 2.

1. The safety significance of pumps and valves whose function is required for safety will be classified as either High Safety Significant (RI-IST High) Components, Medium Safety Significant (RI-IST Medium) Components, or Low Safety Significant (RI-IST-Low) Components. Inservice testing of RI-IST High Components will (nominally) be conducted at the Code-specified frequency using approved Code methods. The inservice testing of those components that have been categorized as RI-IST Medium Components will be performed at extended test frequencies determined in accordance with the RI-IST program description. Additionally, IST Medium Components will be assigned a compensatory measure, as determined in accordance with the RI-IST program description, to assure the continued reliability of the component. The inservice testing of those components that have been categorized as RI-IST Low Components will be performed at extended test frequencies determined in accordance with the RI-IST program description. Unless otherwise specified in the RI-IST program description, inservice test methods for all pumps and valves whose function is important to safety will continue to be performed in accordance with the ASME Code.
2. The safety significance assessment of pumps and valves will be updated every other refueling interval (approximately 3 years) based on Unit 1 refueling, as specified in this report.

This alternative testing strategy will also apply to successive 120-month intervals as discussed in 10 CFR 50.55a(f)(4)(ii).

A review was performed to identify aspects of the plant's design, operation, or other programmatic activities that would be changed by the proposed RI-IST program. No changes are required as a result of the proposed alternative testing strategy. However, since STP will be updating to the 1989 ASME Code, there is a change required to Technical Specification surveillance requirement 4.4.6.2.2.e. This surveillance requirement references paragraph IWV-3427(b) in the 1983 ASME Code for trending leak test results of the reactor coolant pressure boundary isolation valves. The trending requirement is not included in the 1989 ASME Code and an amendment to the STP Technical Specification has been requested in letter NOC-AE-000712.

1.1.1 Basis for Alternative Test Strategy

Current Code-prescribed test intervals are based on a deterministic approach that considers a set of challenges to safety and determines how those challenges should be mitigated. This approach considers elements of probability, such as the selection of accidents to be analyzed as design basis accidents (e.g., the reactor vessel rupture is considered too improbable to be included) and the requirements for emergency core cooling (e.g., redundancy of trains). The alternative testing strategy presented here incorporates a probabilistic approach to regulation that enhances and extends this traditional, deterministic approach by:

- Allowing consideration of a broader set of potential challenges to safety,
- Providing a logical means for prioritizing safety challenges based on risk significance,
- Encouraging the evaluation of current testing strategies and their efficacy in detecting potential challenges to safety, and
- Allowing consideration of a broader set of resources to defend against safety challenges.

First, the PRA model has identified a broader set of challenges to safety. In particular, the RI-IST project team has identified important components that were not in the ASME Section XI IST Program. Even though these components are outside the traditional ASME component eligibility requirements, they will be evaluated to determine if these components are being tested commensurate with their safety significance. If inclusion of the component will reduce plant risk as measured by the change in CDF, then the components will be tested as described below. Where the ASME Section XI testing is practical, the components added to the RI-IST Program will be tested in accordance with the ASME/ANSI 1987 edition of the OM Code with the OMa 1988 Addenda. Where the ASME Section XI testing is not practical or does not apply, alternative methods will be developed to ensure operational readiness.

Second, the RI-IST Testing program prioritizes safety challenges based on the results of the STP PRA, which includes effects from both external event initiators (e.g., flood, tornadoes, fires, and seismic events) and from enhanced common cause failure modeling. The ranking process also considers risk impacts of other operating modes, specifically the most risk-significant plant shutdown configurations. These rankings consider importance with respect to both prevention of core damage and prevention of large early releases of radiation to the public. Section 2 of this engineering analysis describes the methodology used in arriving at RI-IST ranking categorizations.

Third, the RI-IST methodology promotes the evaluation of current testing strategies. If another testing strategy exists (especially for RI-IST High or Medium components), participants will consider whether this new test provides an enhanced understanding of a component's ability to perform its safety function during a design basis accident scenario. Moreover, if the test currently included in the program either tests the function of the component in a nonstandard plant configuration, or places the component(s) involved in the test under increased stresses that, over time, potentially decrease the reliability of the component, then RI-IST participants should endeavor to find an improved testing strategy.

Finally, an IDP allows a broader set of resources to be considered to defend against challenges to safety. The IDP includes a group of experienced individuals with expertise in the areas of ASME Code requirements and testing methodology, plant operations, maintenance, safety analysis engineering, system engineering, design engineering, and probabilistic risk assessment. The IDP ensures that the risk ranking inputs are consistent with plant design, operating procedures, and plant-specific operating experience. More importantly, an integrated decision-making process that incorporates risk insights assures that a defense-in-depth philosophy is maintained (Section 2.4).

1.2 INSERVICE TESTING PROGRAM SCOPE

Aside from exceptions noted in the RI-IST program description contained in Attachment 2, components in the traditional ASME Section XI IST program that are determined to be IST High will continue to be tested in accordance with the current program, which meets the requirements of Section XI of the ASME Boiler and Pressure Vessel Code (except where specific written relief has been granted). Similarly, components in the traditional ASME Section XI IST program which are determined to be IST Low or IST Medium will also be tested in accordance with the ASME Section XI IST program. However, the component's test frequency may initially be extended as detailed in Attachment 2, Program Description Summary. Hence, no components will be removed from the IST program scope. The extended test frequency will be staggered over the respective test interval as described in the RI-IST program description (Attachment 2). The RI-IST program scope for the second 120-month interval includes the valves and pumps listed in tabular reports contained in Attachment 3. The IST Plan document may be found in Attachment 3 of this submittal.

1.3 RI-IST PROGRAM CHANGES AFTER INITIAL APPROVAL

Currently, the risk-informed process has categorized and developed a testing strategy for 1138 of the 1376 STP IST components. As a living process, components will be reassessed periodically as stated in Section 1.1 to reflect changes in plant configuration, component performance, test results, industry experience, and other factors. When significant changes that do not require prior regulatory approval occur, those changes will be provided to the NRC in a program update. All potential future changes will be evaluated against the change mechanisms described in the regulations (e.g., 10CFR50.55a, 10CFR50.59) prior to implementation. Further, any future changes will consider the cumulative risk impact of all RI-IST program changes (i.e., initial approval plus later changes) and the compliance of this calculated risk impact with acceptance guidelines discussed in RG 1.174 and RG 1.175.

2.0 Engineering Analysis

The STP RI-IST project employed a method that blended probabilistic and traditional engineering insights to identify opportunities to reduce those IST-related regulatory requirements and commitments that require significant resources to comply with and/or implement, but contribute insignificantly to safe and reliable operation. The engineering evaluation provides the core information required to support decision-making and risk quantification for a risk-informed IST application of this nature.

The engineering evaluation was divided into the five major tasks listed below:

- Component Function Evaluation
- Component Corrective Maintenance Evaluation
- Calculation of Risk Measures Using the STP PRA
- Component Risk Categorization by Working Group and Review by Expert Panel
- Cumulative Risk Evaluation Using the STP PRA

The component function evaluation established the design basis safety functions of IST components and related these functions to component failure modes modeled by the PRA. Modeling implications were also identified, including the component or system-level assumptions that affect the level of credit the PRA affords an IST component's safety function. The component corrective maintenance evaluation validated the basis for the PRA reliability assessment and demonstrated how it compared to generic and plant-specific experience. It also established a baseline for future monitoring that is needed to compensate for some of the components whose testing requirements are reduced.

The PRA was then used in a variety of ways to evaluate the importance of components and their functions. In this evaluation, calculated risk measures (Section 2.3.2), sensitivity studies, and a cumulative risk evaluation (Section 2.3.3) were used to demonstrate completeness of the risk evaluation. This process was followed by the RI-IST Working Group review and validation of the PRA risk measure, a process that ensured an integrated effort through active technology transfer. The RI-IST Working Group consisted of members with expertise in the areas of power plant operations, plant maintenance, PRA, nuclear safety analysis, systems engineering, design basis engineering, quality assurance, licensing, and Inservice Testing (including ASME B&PV Code Section XI and ASME Code Cases). In addition to considering the basis for the PRA risk measure for modeled components, the RI-IST Working Group qualitatively assessed the following for each component group:

- The degree to which component failure leads to an increase in the frequency of initiating events,
- The degree to which component failure leads to the failure of another safety system,
- The degree to which component failure causes a transient,
- The role of the component in the plant EOPs, and
- The role of the component in plant shutdown.

As part of the process, the RI-IST Working Group authored a narrative basis to support the final RI-IST categorization of each component group.

Subsequent to Working Group initial RI-IST categorization of components, the STP plant Expert Panel (EP) considered and ultimately validated the results of all Working Group activities and studies performed by the IST project members. The Expert Panel consisted of members with expertise in the areas of power plant operations, plant maintenance, PRA, nuclear safety analysis, design basis engineering, and quality assurance. The Expert Panel served as the central point of decision-making for major technical issues and offered guidance to risk-informed IST project members in performing their work.

The strength of this risk-informed IST program and the integrity of its results lie both in the comprehensiveness of the methodology and in the work of both the Working Group and the plant Expert Panel. The IDP presented in Section 2.4 was implemented according to clear guidelines and operated directly from documentation produced in earlier tasks.

Results of the engineering evaluation are discussed in the following subsections.

2.1 LICENSING CONSIDERATIONS

2.1.1 Evaluation of Proposed Changes to Licensing Basis

The risk-informed project team reviewed plant programs to identify STP component-related procedures and programs that credit current IST test intervals. In addition, plant licensing reviewed licensing-related commitments that credit current IST test intervals. No commitments were identified as being adversely affected by the proposed RI-IST program. As part of the RI-IST update, a similar review will be performed to ensure consistency with other plant programs.

Consideration of the original acceptance conditions, criteria, limits, risk significance of the component, diversity, redundancy, defense-in-depth, and other aspects of the General Design Criteria, are addressed by the RI-IST Working Group risk categorization process.

2.1.2 Relief Requests and Technical Specification Changes

Review of existing relief requests, Technical Specifications, and licensee-controlled specifications determined that no new relief requests or exemptions beyond the currently approved relief requests and this submittal are needed to implement the proposed alternative testing strategy and the RI-IST program at this time. However, since STP will be updating to the 1989 ASME Code for the second 120-month interval, there is a change required to Technical Specification 4.4.6.2.2.e. This surveillance requirement references paragraph IWV-3427(b) in the 1983 ASME Code for trending leak test results of the Reactor Coolant pressure boundary isolation valves. The trending requirement is not included in the 1989 ASME Code and an amendment to the STP Technical Specification removing the requirement has been requested in letter NOC-AE-000712.

STP does not plan to resubmit previously approved relief requests for components ranked as RI-IST High, as the existing relief requests were evaluated as part of the Working Group deliberations and were therefore incorporated into the decision-making process. However, Cold Shutdown Justifications, Refueling Outage Justifications, and approved Relief Requests are shown in Attachment 3 of this submittal as a part of the second 120-month interval IST Plan.

This submittal requires no new relief requests or exemptions beyond those currently approved for either risk categorization, as the program implementation plan contained in Attachment 2 does not seek to extend the test intervals for these components more than is allowed by Regulatory Guide 1.175. Therefore, these components will continue to be tested at their Code-prescribed intervals, unless justified based on plant conditions required for testing.

STP's RI-IST program results in the testing of RI-IST High components in accordance with the Code test frequency and method requirements or enhanced test methods and corresponding frequencies that have been previously approved. Similarly, STP will test RI-IST Low and RI-IST Medium components in accordance with the Code test method requirements (although at an extended interval) or using previously approved enhanced testing methods and corresponding frequencies. STP concludes that additional relief requests are not required to implement test methods that are in accordance with ASME Code requirements or ASME Code Cases approved by the NRC.

For the high risk significant components that are not within the scope of the current IST program, it is not practicable to perform Code testing. However, as these components are highly safety significant, STP is considering the efficacy and practicality of either adding these components to the RI-IST program, or adding RI-IST monitoring and trending to ensure their continued operability. Section 2.3.2.2 discusses these components and the plant activities already being performed to ensure their continued reliability.

2.2 TRADITIONAL ENGINEERING EVALUATION

This part of the evaluation utilizes traditional engineering methods to evaluate the potential effect of the proposed RI-IST program on defense-in-depth attributes and safety margins. Because of its importance to reactor safety and to the health and safety of the public, the concept of defense-in-depth is considered to be one of the key safety principles to be addressed by any risk-informed application. The maintenance of safety margins is also a very important part of ensuring continued reactor safety and is included in the list of key safety principles to consider.

2.2.1 Defense-in-Depth Evaluation

The STP RI-IST program has been developed consistent with the RG 1.174 guidelines for maintaining defense-in-depth. RG 1.174 lists seven acceptance guidelines for determining whether defense-in-depth has been addressed adequately by a risk-informed program:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences of challenges to the system (e.g., no risk outliers).
- Defenses against potential common cause failures are preserved and the potential for introduction

of new common cause failure mechanisms is assessed.

- Independence of barriers is not degraded.
- Defenses against human errors are preserved.
- The intent of the General Design Criteria in 10 CFR Part 50, Appendix A is maintained.

The following indicates how the STP RI-IST program specifically meets this definition of defense-in-depth. Finally, this section discusses how the use of multiple PRA importance measures and the complementary risk metrics of CDF and large early release frequency (LERF) provide additional assurance that defense-in-depth is maintained.

A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.

The use of multiple risk metrics, including CDF and LERF, ensures a reasonable balance between risk prevention methods (e.g., testing strategies). The basis for this statement is provided in further detail in Section 2.2.1.1.

The STP RI-IST program results further demonstrate that such a reasonable balance exists. The components whose failure can most affect that balance are categorized as RI-IST High. For example, important steam generator tube rupture containment isolation valves (CIVs) are among the components categorized as IST High. It is these components whose failure can not only contribute to the loss of core cooling, but can also cause containment failure and limit the effectiveness of consequence mitigation.

The STP RI-IST program actually improves the balance in prevention methods (e.g., testing strategies) by adjusting the IST program to further enhance safety. Specifically, the RI-IST program reduces unintended adverse impacts of ISTs on components by replacing the current LHSI pump test with a full flow test.

Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.

The STP RI-IST does not introduce reliance on new programmatic activities. The compensatory measures used to ensure that degradations in equipment performance can be quickly detected are chosen from either normal plant operational activities (e.g., swapping the trains in operation) or existing preventative maintenance activities, both of which are existing plant program elements. These compensatory measures help to more clearly communicate which plant programmatic actions are important to ensure that uncertainties in equipment performance are minimized.

System redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences of challenges to the system (e.g., no risk outliers).

The preservation of system redundancy, independence, and diversity is a natural outcome of PRA if the plant risk profile contains a balance of core damage risk sources. The IDP process can ensure these conditions are met by understanding the reasons why components are categorized as RI-IST High, RI-IST Medium, or RI-IST Low.

The STP PRA models a balance in sources of core damage risk. The sources of risk in turn include severe accidents that result from design basis accident initiators such as large break loss of coolant accidents (LOCAs) and steam generator tube ruptures. The balance in risk causes the categorization of components using PRA to be done on an evenhanded basis covering the full scope of safety functions.

The STP risk profile includes important risk considerations from a wide spectrum of sources. Stated simply, risk is relatively well balanced. There are important risk contributions from internal event initiators as well as location-dependent, external event initiators. For example, besides station blackout and other internal event risk sources, location dependent risk sources such as flood play important roles. In the internal event sources, contributions from transients, support system failures, offsite power interruptions, Anticipated Transient Without Scram (ATWS), LOCAs, and steam generator tube ruptures all make contributions to the risk profile.

As a result, the components which mitigate the spectrum of accidents are not ranked low solely because of initiating event frequency. Further, sensitivity studies performed for human actions ensure that components which mitigate the spectrum of accidents are not ranked low solely because of the reliability of a human action. The implication of these findings is that uncertainty in initiating events or human errors does not play an important role in component categorization. In addition, no single safety function was found to be insignificant, a situation that would have caused all components within that function to be insignificant. For example, the safety functions that uniquely mitigate LOCAs, provide reactivity control, and mitigate steam generator tube ruptures all make important contributions to the risk profile. Thus for STP, components which support these functions are represented in the risk profile.

After selecting numerical importance criteria and applying them to the components, the RI-IST Working Group and Expert Panel developed an understanding of the basic reasons why components were categorized RI-IST High, RI-IST Medium, or RI-IST Low. This effort included reviewing importance measures in the P&ID format and understanding the way that component reliability and redundancy impact component categorization. This understanding was a fundamental part of the Integrated Decision-making Process.

When the component categorization method is applied to IST pumps and valves using a PRA whose sources of risk are well balanced, the following observations can be made.

Observation number 1: The level of redundancy within each safety function greatly influences component categorization. Table 2.2-1 indicates how participants in the integrated decision-making used the concept of "average redundancy" in the STP plant design to draw conclusions regarding component categorization.

Table 2.2-1: Relationship of Defense-in-Depth to Component Categorization

DEGREE OF REDUNDANCY	CLASSIFICATION TO ENSURE DEFENSE-IN-DEPTH IS MAINTAINED	ADDITIONAL RESTRICTIONS
Less than average redundancy	all components assigned RI-IST High	N/A
Average redundancy	Assigned RI-IST Medium; only reliable components are treated like RI-IST Low provided these components are assigned a compensatory measure	poorly performing components classified as RI-IST High, components important to CCF classified as RI-IST High
Greater than average redundancy	typical treatment for RI-IST Low components	poorly performing components classified as RI-IST High, components important to CCF classified as RI-IST High

As the table shows, the most restrictive aspects of the RI-IST program apply to those elements with the least amount of redundancy. Relaxation in the STP RI-IST program occurs only when the relative level of redundancy is increased. The highest level of relaxation occurs only when there is greater than average redundancy.

However, merely having multiple trains of a component available in a system does not automatically result in a lower risk categorization for a component. When considering whether component redundancy or diversity is a factor, the RI-IST methodology evaluates redundancy based on system operating configuration, reliability history, recovery time available, and other factors. The process necessitates an examination of the effect of the component failure on each system function supported by that component. The primary consideration is whether failure of the component will fail or severely degrade the function. If that is not the case, then participants may factor in component redundancy, as long as the component's reliability and that of its redundant counterpart have been satisfactory.

In addition to ensuring redundancy is preserved, the STP method also ensures that diversity is maintained. Again, this outcome depends on the well-balanced nature of risk and some specific attributes (redundancy and reliability) as the IDP process confirmed it.

Observation number 2: A system that has less diversity is more subject to CCF. Said another way, when like components (i.e., not diverse) can cause failure of the system, common cause methods predict an increased CCF contribution. When more diverse components are included, for example a mixture of turbine-driven and motor-driven pumps, the CCF contribution is lower.

The Expert Panel concluded that components that had significant contributions to CCF were RI-IST High components. This action had the effect of avoiding relaxation of requirements on those components with the lowest level of diversity within the system.

Defenses against potential common cause failure are preserved and the potential for introduction of new common cause failure mechanisms is assessed.

The preservation of defenses against CCF is partially addressed above when it is indicated that components important to CCF are ranked RI-IST High. More importantly however, the implementation and monitoring method discussed in the RI-IST Program Description (Attachment 3) both preserve defenses and ensure that potential increases in CCF are quickly detected. Regarding implementation, staggering of testing provides additional assurance against CCFs. Regarding monitoring, the STP Condition Reporting Process investigates failures to determine if the potential exists for like component failures.

Independence of barriers is not degraded.

The multiple barriers to loss of core cooling, containment integrity and release mitigation are preserved as described above. No new dependencies are introduced and the potential for CCF across barriers is minimized by the approach to implementation and monitoring.

Defenses against human errors are preserved.

The sensitivity studies for the human reliability analysis show no changes to component categorization. During development of the program, no procedure changes were made to increase the reliance on operator actions. Probably most important, by reducing the number of ISTs and therefore, requiring less off-normal alignments to perform them, operator burden is reduced by the RI-IST program. Finally, Operations' input is a key part of the integrated decision-making process.

The intent of 10CFR50 Appendix A is maintained.

When the PRA does not explicitly model a component, function, or mode of operation, a qualitative method is used to classify the component as RI-IST High, RI-IST Medium, or RI-IST Low and to determine whether a compensatory measure is required to assure the continued reliability of the component. The qualitative method is consistent with the principle of defense-in-depth because it preserves the distinction between those components that have high relative redundancy and those that have only high relative reliability.

The STP RI-IST program does not eliminate ISTs in any safety function. It does, however, change the interval of ISTs. When the basis for the change in interval is reliable equipment performance, compensatory measures are used to ensure the performance is well known and that timely feedback of operational performance will occur.

These efforts ensure that the intent of GDC 10 CFR 50 Appendix A is maintained by applying key safety principles (regardless of whether the PRA explicitly models the component), and by not eliminating ISTs.

2.2.1.1 The Use of Multiple Risk Metrics to Ensure Defense-in-Depth

The following describes how the use of multiple risk metrics, namely CDF and LERF, provides an initial basis for ensuring defense-in-depth. The traditional defense-in-depth concept as used in the STP UFSAR is to maintain multiple barriers that restrict or limit the transport of radioactive material from the nuclear fuel to the public. These barriers are:

- Fuel pellet matrix
- Cladding
- Reactor Coolant System (RCS)
- Containment building

PRAs analyze the integrity of all these barriers, although the first two tend to be implicitly modeled and the last two explicitly modeled. CDF is a measure of the first three barriers. The containment building integrity is measured in terms of LERF. As long as these two parameters (i.e., CDF and LERF) are maintained at reasonably low frequencies, then it should be concluded that these two barriers (i.e., reactor coolant system and containment building) are most likely capable of performing their functions, when needed. This, in turn, means that the defense-in-depth capabilities are well controlled and maintained.

CDF:

The STP RI-IST program used Fussell-Vesely (FV) and Risk Achievement Worth (RAW) importance measures to initially prioritize the IST components based on their risk significance. Since these two importance measures may have some limitations, various sensitivity studies were conducted along with other considerations to ensure the completeness of the approach.

When a nuclear plant has an acceptable CDF, it means that plant components are reliable and/or there is enough redundant equipment available to perform the required accident mitigating function when needed. The redundancy could be at the component level, train level, system level, or function level. For example, at the function level, if all trains of the Auxiliary Feedwater system (AF) fail, the secondary heat transfer function will be lost. All components necessary to provide the AF flow path function are included in the RI-IST High category. For other functions with more redundancy, fewer components are included in the RI-IST High category but an equal or greater measure of safety is maintained.

Therefore, the STP ranking results demonstrate that, in effect, defense-in-depth is inherently assured. If the risk importance values of the IST components have been properly evaluated, and sufficient sensitivity studies have been performed[10], and their cumulative impact on total CDF has been calculated to be low, and the resulting CDF is still low, then there are still adequate redundancies at different levels available to mitigate the consequences of a severe accident. This, in turn, leads to the fact that the defense-in-depth capabilities are adequately maintained even with all the proposed changes to the test intervals of the low-ranked components. In addition, testing and maintenance strategies that assure the reliability of

[10] The RI-IST program study employs the results of the risk-informed GQA program study.

components will be either maintained or optimized in the proposed RI-IST program.

LERF:

The same risk importance approach used for CDF was applied to LERF. Similar sensitivity studies[10] were conducted to compensate for the limitations of FV and RAW importance measure techniques. In addition, in order to ensure that the containment integrity is always maintained, the following issues were also considered in the study:

- Containment isolation features that may not directly impact the value of LERF.
- Interfacing systems LOCA that provides a direct release path to the outside containment.

Furthermore, similar to the CDF impact evaluation, another study was performed to evaluate the cumulative impact of the requested changes to the current IST program on total LERF. The results of this study for STP demonstrated that modifying the test frequencies of the IST components in the less safety significant category to every 54 months is reasonable. When total LERF is low, it means that containment safeguards features are reliable and/or there are enough redundant components available to perform similar functions, when required. This leads to the fact that the defense-in-depth capabilities are adequately maintained with the proposed changes to the test intervals of the low-ranked components.

2.2.2 Safety Margin Evaluation

The STP RI-IST program assures that sufficient safety margin is maintained. The basis for this conclusion is that the RI-IST program merely extends the test interval for certain IST components. For these interval extensions, corresponding program actions to monitor component performance are taken to ensure the overall safety margin does not degrade. (Refer to the Performance Monitoring and Feedback And Corrective Action discussions in the RI-IST Program Description, Attachment 2.) Further, the RI-IST program does not seek to reduce the scope of the IST program. Safety analysis acceptance criteria (e.g., UFSAR, supporting analyses) will continue to be met as before.

In fact, the RI-IST program considers increases to the IST program scope. The RI-IST program does not remove any components from the current IST program; however, it considers adding highly risk significant components, such as dampers, that are outside traditional Code class boundaries. Additionally, the program does not remove any safety functions. It builds an awareness of risk functions by identifying them side by side with safety functions. Finally, there are no degradations in the effectiveness of test methods. Indeed, this program proposes to enhance test methods, in particular that associated with the LHSI pumps. Consequently, these program improvements should tangibly enhance the safety margin.

In addition to tangible scope enhancements, the safety margin is also enhanced because the RI-IST program includes three changes that should improve the understanding of component performance:

(1) For RI-IST Medium components, the program includes compensatory measures that are effective fault finding tasks. The observed performance during these fault-finding tasks is now linked directly to the IST program performance, providing a more integrated view of safety margin and

the ways that different plant programs affect and monitor it.

(2) The program uses a phased implementation approach so that a change in performance of structures, systems and components (SSCs) resulting from extending the interval can be identified and fed back to the program via the plant-wide corrective action program (i.e., STP's Condition Reporting Process). This improved understanding of how component performance relates to test interval may provide insights that in turn could even improve the process for maintaining the design margin of RI-IST High components.

(3) There are PRA-important components not in the current IST program (ASME and non-ASME components) that are potential long-term additions to the program (e.g., pumps, chillers, fans, and dampers). Not only could this potentially reduce the overall CDF, but it will also provide insight into the value of IST programs in maintaining and improving component margin. That is, the change in performance and margin can be measured for the case when a component is brought into the IST program.

When these three items are taken together with component performance changes from enhanced test methods, the uncertainty associated with component failure rates as a function of time should be reduced. This reduction in uncertainty should further improve safety margins.

The proposed RI-IST program will improve RI-IST High component availability and ensure that changes to the reliability of RI-IST Medium and RI-IST Low components will not be significant. Overall, as discussed in Section 2.3.3, the RI-IST program will be safety neutral.

2.3 PROBABILISTIC RISK ASSESSMENT

The PRA study for STP fully satisfies the requirements of a full-scope level 2 PRA and includes the effects of external events and fires. The PRA was primarily developed to support changes to the plant technical specifications to allow full credit for the plant's unique three-train design.

One of the main objectives of the PRA development was to be able to utilize its results and insights toward the enhancement of plant safety through risk-informed applications. With this objective in mind, the PRA elements were developed in detail and integrated in a manner sufficient to satisfy both the NRC Generic Letter 88-20 requirements and support future plant applications, such as the risk-informed application evaluated in this report.

The STP RI-IST program presented in this submittal meets the objectives outlined in the Commission's PRA Policy Statement in that the evaluation demonstrates that the proposed changes do not compromise the principles of defense in depth, nor do they degrade safety margins.

2.3.1 Scope, Level of Detail, and Quality of the PRA for RI-IST Application

2.3.1.1 PRA Scope

The original STP PRA model was a level 1 analysis that included a full range of external events, including detailed fire analysis. This model was completed about the same time that Generic Letter 88-20 was issued. The level 1 model was submitted for NRC review to support proposed technical specification changes while a level 2 model was developed in order to satisfy the Generic Letter requirement. The final IPE was submitted in 1992[11]. The SER for the level 1 PRA is documented in NUREG/CR-5606[12]. The NRC acceptance of the external events analysis is documented in a letter dated December 15, 1998[13]. Additional reviews of the STP PRA have been performed to support subsequent technical specification changes and the Graded Quality Assurance Program[14,15].

The current STP PRA, documented as STP_1997[16], includes all external events and is a complete level 2 analysis of core damage frequency and large early release frequency of the South Texas Project Electric Generating Station. Some of the external events that are addressed in the STP PRA include:

- External floods from main cooling reservoir breach,
- Tornado that fails offsite power and the essential cooling pond,
- Seismic events from 0.1 to 0.6g[17], and
- Internal fires.

The evaluation of seismic events and other external events are well beyond the design basis external events. All of these external events are included in the STP PRA results and are explicitly included in all risk categorizations that are based on the PRA.

In addition, the PRA accounts for common cause failures of all active components. STP believes the proposed methodology of dividing the common cause importance value into the individual elements is an innovative approach and is a more technically correct method to account for common cause within a single importance measure. However, due to issues associated with this methodology and the time necessary to gain consensus on this approach, the STP PRA has reverted to the recognized approach for PRA risk rankings from the GQA SER[6].

Reverting to the GQA SER common cause methodology is documented and tracked under STP's corrective action program. The corrective actions to address this condition include the following activities:

1. Revising the risk ranking analysis, and
2. Identifying components requiring re-categorization.

PRA representatives have completed this analysis for IST components and have identified those

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[11] NRC's (Office of Nuclear Reactor Regulation) January 21, 1992 safety evaluation report on the Level I PSA submitted on April 14, 1989.

[12] NRC's (Office of Nuclear Reactor Regulation) August 31, 1993 safety evaluation on the external events analysis in the Level 1 PSA submitted on April 14, 1989.

[13] NRC's (Office of Nuclear Regulatory Research) June 27, 1995 staff evaluation of the Level 2 enhancements made to the 1989 PSA and submitted as the licensee's Individual Plant Examination (IPE) on August 28, 1992.

[14] South Texas Project Electric Generating Station Level 2 Probabilistic Safety Assessment and Individual Plant Examination, August 1992.

[15] A Review of the South Texas Project Probabilistic Safety Analysis for Accident Frequency Estimates and Containment Binning, NUREG/CR-5606, August 1991.

[16] Review of South Texas Project Units 1 and 2 Individual Plant Examination of External Events (IPEEE) Submittal NRC letter, dated 12/15/98.

components affected by this decision. Affected components have been conservatively shifted from lower risk categorizations to RI-IST High, signifying that these components will not be eligible for test interval extension.

Finally, the PRA includes planned and unplanned maintenance configurations, and test configurations that affect train line-up or operability. The model reflects the as-built and as-maintained plant and is consistent with the definition of a full-scope model described in RG 1.174. The model supports the STP-developed on-line risk monitor, RAsCal[18], which is used to control on-line maintenance at STP.

With respect to the scope of the specific IST components modeled by the PRA, pumps and valves that are important to systems required to prevent core damage and radioactivity release are explicitly modeled. Categorization of the risk significance of the modeled equipment is based on risk importance metrics generated from this full scope PRA, integrated with the deterministic knowledge of the RI-IST Working Group. Pumps and valves that are in the In-Service Testing Program, but are not modeled in the PRA have been categorized by the RI-IST Working Group, which considered the following factors when determining the categorization of each IST component:

- Core damage frequency,
- Radioactivity release prevention,
- Level of redundancy,
- Operational requirements,
- Use in the plant emergency procedures,
- Shutdown configurations, and
- Prevention of a plant initiating event.

2.3.1.2 Level of Detail

The STP PRA models the specific failure modes of the pumps and valves. In some cases, the pumps and valves have more than one failure mode. For valves, these failure modes may include failure to open, failure to close, failure to operate, failure on demand (open or reseating), or failure to transfer to the failed position. For pumps, the PRA models failure to start and failure to run. Mapping of these failure modes to the associated component permits calculation of component-specific FV and RAW importance values, which is consistent with the requirements of RG 1.174. Given mapping of this nature, this full-scale application of the PRA establishes a cause-effect relationship that identifies the portions of the PRA affected by a proposed test interval extension. Therefore, the level of detail of the PRA supports a completely quantitative analysis of the impact of proposed test interval extensions on plant risk.

2.3.1.3 PRA Quality

STP has a level 1/level 2 PRA which includes external events. The external events portion contains both

[17] The safe shutdown earthquake for STP is 0.1g.

[18] Notice of Consideration of Issuance of Amendments - South Texas Project, Units 1 and 2 (Tac Nos. M92169 and M92170), Safety Evaluation Report of Diesel Generator Extended Allowed Outage Time, NRC letter dated February 2, 1996.

a Fire PRA (with Spatial Interactions analysis) and Seismic PRA analysis. The STP PRA has been structured to have a comprehensive treatment of common cause failures and plant configurations. A detailed human reliability analysis is also included.

Previous Reviews

Results of reviews of the STP PRA are documented by the following:

- "A Review of the South Texas Probabilistic Safety Analysis for Accident Frequency Estimates and Containment Binning" contracted through Sandia National Laboratories. NUREG/CR 5606;
- "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to the Probabilistic Safety Analysis Evaluation," sent to the Houston Lighting & Power Company under cover letter dated January 21, 1992;
- "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to the Probabilistic Safety Assessment - External Events," sent to the Houston Lighting & Power Company under cover letter dated August 31, 1993;
- "Issuance of Amendment Nos. 59 and 47 to Facility Operating License Nos. NPF-76 and NPF-80 and Related Relief Requests – South Texas Project, Units 1 and 2 (TAC Nos. M76048 and M76049)" sent to Houston Lighting & Power Company February 17, 1994;
- "Individual Plant Examination (IPE) - Internal Events, South Texas Project, Units 1 And 2-(STP) (TAC Nos. M74471 and M74472)" dated August 9, 1995 (Included equipment survivability analysis);
- "South Texas Project, Units 1 and 2 – Amendment Nos. 85 and 72 to Facility Operating License Nos. NPF-76 and NPF-80 (TAC Nos. M92169 and M92170)" sent to Houston Lighting and Power Company under a cover letter dated October 31, 1996. This amendment allows extension of the standby diesel generator allowed outage time to 14 days, and extension of the essential cooling water and essential chilled water allowed outage time to 7 days;
- "Graded Quality Assurance, Operations Quality Assurance Plan (Revision 13), South Texas Project, Units 1 and 2 (STP)(TAC Nos. M92450 and M92451) dated November 6, 1997.

PRA Maintenance

STP's PRA Configuration and Control program is structured to ensure changes in plant design and equipment performance are reflected in the PRA as appropriate. The PRA Configuration and Control process is administered by procedures and guidelines that ensure proper control of all changes to the models by persons independent from the person making the change and approved by the PRA supervisor. STP's PRA will undergo a PRA certification under the Westinghouse Owner's Group Peer Review Process[19] and is expected to be in compliance with the ASME PRA standard for risk-informed

[19] The Westinghouse Owners Group (WOG) Certification of the South Texas Project PRA is currently scheduled for April 2002

applications.

PRA Self-Assessment

A self-assessment of the overall control process was performed using the guidance from the BWR Owner's Group Peer Certification Process. All findings from this self-assessment were documented in the corrective action program and have been corrected. The conclusions from the self-assessment indicate that the methods used to control the PRA satisfy the appropriate requirements of Appendix B to 10CFR50. Given the current state-of-the-art in PRA analyses and techniques, as well as the control of the processes used to make changes to the model, the quality of the PRA is sufficient to achieve reliable results for this relief request.

In summary, the STP PRA has been subjected to extensive peer and regulatory review. The PRA model, assumptions, database changes and improvements, and computer code are controlled and documented by administrative procedure. The model and database reflect the as-built plan and the most recent historical data. Finally, in its review of the PRA in support of STP's request to implement a graded quality assurance (GQA) program, the staff stated that the process STP intends to use to maintain the PRA and to evaluate future risk changes is adequate, and that, "...on the basis of this review, [the staff finds that] the quality of the PRA analysis, which includes the PRA models and the various application specific bounding studies, is sufficient for the assigning of SSCs (in relation to their importance to the CDF and LERF metrics) into broad safety-significance categories. In addition, the staff finds that the PRA assumptions and SSC categories are sufficiently well defined."[20] Therefore, the STP PRA is of a quality consistent with that required to perform accurate, thorough, and comprehensive evaluations for a risk-informed IST application.

2.3.2 Categorization of Components

This section provides a more detailed description of the technical details which support the component categorization process used for the STP RI-IST program, with emphasis placed on issues that were addressed to successfully implement the process, as well as the risk ranking results.

The STP RI-IST program implemented the same methodology that was applied in recent years during other risk-informed efforts at STP, including the NRC-approved GQA program[6] and the recently-submitted request for exemption from special treatment requirements[7]. As was indicated in the NRC SER for the GQA program, "...the staff finds that the importance measures calculated by the licensee, and the guidelines used to develop the PRA-based categorization from these measures, are reasonable and consistent."[20] The major exception to the GQA ranking process was the elimination of passive failures for the components included in the IST program. The IST program as implemented does not test for passive failure modes of components (i.e., the IST does not perform test activities aimed at verifying that components remain in safety positions).

[20] "Safety Evaluation by the Office of Nuclear Reactor Regulation [Related to the] Houston Lighting and Power Company South Texas Project, Units 1 and 2, Graded Quality Assurance Program, Docket Numbers 50-498 and 50-499," section 3.2.6.

The development of risk importance measures for ranking required selecting the measures to be used, selecting the number of categories and ranges for each importance measure, and determining the implication of each category to inservice testing. This risk-informed application employed the FV and the RAW probabilistic risk importance measures. Because the RI-IST initiative endeavors to reduce existing regulatory burden rather than focus on new regulatory initiatives, this methodology applies these risk measures in a manner intended to ensure a safety neutral outcome.

Fussell-Vesely provides a measure of incremental change in total CDF that indicates the importance of incremental changes in reliability that might result from changing inservice test intervals. Risk Achievement Worth provides an indicator of the importance of degradations in component reliability and is, in essence, a measure of functional importance. That is, two components having the same functional role, e.g., in the same "functional train", will have the same RAW. Risk ranking results generally indicated that such functionally similar components could have sufficiently different Fussell-Vesely measures. Often the differences were such that one could be ranked high and another low. This finding implies that the analyst must be relatively certain of a component's failure probability to draw reliable insights from the FV measure.

These measures were combined into the component categorization decision criteria described in the following table:

PRA RANKING	**CRITERIA**
High	RAW ≥ 100.0 or FV ≥ 0.01 or FV ≥ 0.005 and RAW ≥ 2.0
Medium (Further Evaluation is Required)	FV < 0.005 and 100.0 > RAW ≥ 10.0
Medium	0.01 > FV ≥ 0.005 and RAW < 2.0 or FV < 0.005 and 10.0 > RAW ≥ 2.0
Low	FV < 0.005 and RAW < 2.0

As the table indicates, components with a significant FV (FV ≥ 0.01, or FV ≥ 0.005 when RAW is also ≥ 2.0) and/or RAW (RAW ≥ 100.0) were considered "highly risk significant". Components with an insignificant FV (FV < 0.005) were considered "less risk significant". However, it was important to ensure that a reduction in test intervals did not allow unintended consequences, i.e., a compromise in safety resulting from a degradation in reliability. Therefore, the ranking process adapted the RAW to compensate for the weakness in the FV measure. If FV was insignificant (FV < 0.005), it was also required that RAW be small (2.0 < RAW < 10.0), or the RAW had to be insignificant (RAW ≤ 2.0) if the FV were greater than the "insignificant" threshold (FV ≥ 0.005) for a component to be classified as "less risk significant". If RAW was significant, the component was considered by the Working Group for placement in the high category. If the Working Group decided the component could be ranked low, an additional requirement was imposed before a component could be classified as "less risk significant". A compensatory measure was required to be selected by the Working Group to limit degradations in reliability. For the purposes of this study, a compensatory measure is an equivalent stroke of the valve or the equivalent pump start.

Ranking Thresholds

The IST components were divided into three importance categories based on the risk metrics discussed above, FV and RAW. Metric thresholds were chosen such that completeness issues were addressed, and such that each category is accompanied by distinct test requirements. The risk thresholds established for the purposes of component categorization relied upon engineering judgement and were based on a three-category structure according to the following criteria:

CATEGORY	CRITERIA	TEST REQUIREMENTS
RI-IST High	RAW ≥ 100.0 or FV ≥ 0.01 or FV ≥ 0.005 and RAW ≥ 2.0	Current Code-prescribed test(s) or enhanced test(s)
RI-IST Medium	0.01 > FV ≥ 0.005 and RAW < 2.0 or FV < 0.005 and 10.0 > RAW ≥ 2.0	Current Code-prescribed test(s) or enhanced tests if practicable, relaxed test interval (based upon staggered testing model), Compensatory measure as practicable
RI-IST Low	FV < 0.005 and RAW < 2.0	Current Code-prescribed test(s), relaxed test interval (based upon staggered testing model)

In general, the Working Group agreed with the risk categorization suggested by the FV and RAW ranking criteria discussed in the above table. As a matter of process, the RI-IST Working Group considers several component attributes --system operating configuration, reliability history, recovery time available, and other factors--when assigning an overall RI-IST ranking categorization. Regardless, per the STP Comprehensive Risk Management Program (CRMP), 0PGP02-ZA-0003, in all cases, a component's final categorization cannot be lower than the risk categorization based on PRA information if the component is explicitly modeled in the PRA. After the RI-IST Working Group completed its component categorization effort, the Expert Panel reviewed the preliminary results. As a result of the Expert Panel review, the risk ranking for several components was revised to ensure consistency with risk-rankings developed to support the GQA Program.

The ranking criteria established for the STP RI-IST program were found to be practical to implement, generally consistent with the deterministic insights of the Working Group and plant Expert Panel, and effective in producing a safety neutral outcome. Section 2.3.3 contains a discussion of the cumulative risk impact of extending test intervals for RI-IST Medium and RI-IST Low components according to the ranking guidelines suggested by the above criteria.

Results of Component Categorization

A correct application of the component categorization technique described above depends on comparing and establishing a clear relationship between the component function tested within the IST program tests and that function modeled in the PRA.

The initial risk importance determination was performed using the at-power PRA, which includes the effects of both internal and external initiating events, and of common cause modeling. The ranking methods described above were used to establish preliminary component rankings for modeled components. The IDP component ranking categorization, which considers the results of the risk measure calculations at the component level, are contained in a report titled, "RI-IST Component Categorizations and Test Frequencies," which is part of Attachment 3 of this submittal.

The final results of the IDP ranking process are shown below:

RISK RANKING	PERCENTAGE OF COMPONENTS[21] (UNITS 1)
RI-IST High	10.3% (56 components)
RI-IST Medium	15.5% (84 components)
RI-IST Low	69.2% (375 components)
Components with only Appendix J testing(will be dealt with under Appendix J, Option B)	5% (27 components)

Of the components considered for risk categorization, 84.7% (includes both the RI-IST Low components and the RI-IST Medium components) are eligible for interval extension. The remaining IST components – including 90 new relief valves and skid-mounted valves, such as those in the Diesel Generator system—will not be categorized at this time. Instead, they will continue to be tested at the current Code-prescribed test intervals.

Effects of External Events on Component Categorization

The effects of external event initiators (which include fire, external flood, high winds, and seismic events) on the IST components modeled by the PRA did not shift the importance of components. STP has recently provided the NRC with estimates of SSC importance for different categories of external events. The estimates were developed for fires, floods, and seismic initiating events. A full quantification of the PRA model was performed for each calculation of the external event importance measures. The same PRA ranking methodology used to calculate the composite component importance was used for these

21 Containment isolation valves to be tested per 10 CFR 50, Appendix J, Option B account for less than 5% (27 components) of the Unit 1 IST components.

studies.

STP reported that for each case, the component's risk rank resulting from the external event calculations was never higher than the composite PRA risk rank. In other words, no component increased in risk rank category when only the external event categories were analyzed. In general, fires, floods, and seismic events guarantee failure of affected components. Components failed by external events do not influence the mitigation of accident/transient events and have no calculated importance measures. Based on its evaluation, STP concluded that its PRA risk ranking process is not sensitive to the influence of external events and that it appropriately factors in the impacts of external events.

Effects of Common Cause Failure on Component Categorization

Common cause failure is included in the STP PRA for all active components. The common cause method uses the Multiple Greek Letter (MGL) model. The MGL terms are updated on the same frequency as other plant-specific database variables. The FV and RAW risk importance measures include the rank of the associated common cause terms in the determination of all basic event importance measures. Moreover, during the RI-IST Working Group meetings, members deterministically addressed the issue of common cause to ensure that the final component categorization adequately considers the effects of common case failures.

Inclusion of CCF modeling in the at-power risk metrics further affected the risk categorization of IST components. The Expert Panel shifted the rank of 25 check valves in each unit from lower RI-IST ranking categories to higher categories based solely on inclusion of CCF basic events in the RAW risk metric. The following table shows the valve groups that changed ranking categorizations once revised CCF impacts were included in the risk metrics:

GROUP	GROUP DESCRIPTION
AF01	Auxiliary Feedwater Supply to Steam Generator Inside Containment Isolation Check Valves
AF07	Auxiliary Feedwater Auto Recirculation Valves
CC29	CCW Supply to RHR Pump and Heat Exchanger Inside Containment Isolation Check Valve (Trains A, B, and C)
EW08	Essential Cooling Water Pump Discharge Check Valve (Trains A, B, and C)
RH06	Residual Heat Removal Pump Discharge Check Valves (Trains A, B, and C)
SI18	High Head Safety Injection Pump Discharge Inside Containment Isolation Valves (Trains A, B, and C)
SI19	High Head Safety Injection Pump Discharge Check to Cold Leg (Class 1 Boundary) (Trains A, B, and C)
SI21	Low Head Safety Injection Pump Discharge Inside Containment Isolation Valves (Trains A, B, and C)
SI23	Accumulator to Cold Leg Inboard Check Valves (Trains A, B, and C)
SI25	Safety Injection Pumps Suction Check Valves (Trains A, B, and C)

This is a more conservative approach than ranking each component based upon its independent event and subsequently looking at common cause as a sensitivity study. The result is that more components affecting PRA are ranked as RI-IST High, with fewer components ranked as RI-IST Medium or RI-IST Low.

<u>Effects of Shutdown Configurations on Component Categorization</u>

The STP PRA does not yet extend to refueling/shutdown conditions. However, STP currently uses an outage tracking tool (ORAM/Sentinel) to provide useful insights into plant risk during shutdown conditions. The RI-IST Working Group explicitly considered the role of each component in shutdown scenarios and deterministically assessed how the failure of the component to perform its safety function would impact the ability of plant operators to achieve and maintain safe shutdown. For example, the RI-IST Working Group indicated that failure of the Main Steam power-operated relief valves (PORVs, RI-IST group MS03) did have a dominant role in achieving safe shutdown. The PORVs must open to remove decay heat. PRA credits the opening of one of four available PORVs. If the PORVs fail to open, there are twenty available safety valves that can help remove decay heat. The ability to remove decay heat is extremely important; hence, the plant is designed with several available flow paths to provide decay heat removal. Nevertheless, to achieve safe shutdown, this function is particularly important. Therefore, the Working Group indicated this in its narrative basis, and in so doing, they elevated the importance of the PORVs.

As a result of the RI-IST Working Group review, no component groups shifted categories from RI-IST Low or RI-IST Medium to RI-IST High based solely on the impact of component failure on achieving or maintaining safe shutdown. However, as the above example illustrates, shutdown risk scenarios were adequately considered during the component categorization process, especially for those components that provide required boron injection capability during shutdown.

Summary

The purpose of ranking IST components according to their importance lay in assigning specific testing requirements according to safety significance. In order to achieve a safety neutral outcome, the process for component categorization must be scrutable. The preceding discussion demonstrates that this is indeed the case for this risk-informed application.

The following sections further describe the methodology and results, providing additional detail to facilitate a more in-depth understanding of the body of this RI-IST effort. Specifically, important quantitative and qualitative aspects of the probabilistic risk assessment are addressed, followed by discussions of the completeness and adequacy of the risk models. A thorough treatment of the cumulative impact of extending inservice test intervals of RI-IST Medium and RI-IST Low components on plant risk is also included. This discourse provides technical justification for proposed test intervals for less risk significant components in the existing IST and demonstrates how these risk impacts compare to the quantitative CDF and LERF risk increases specified in RG 1.174. Finally, a review of the integrated decision-making process demonstrates the RI-IST Working Group and Expert Panel members' knowledge of plant risk, plant design, plant operations, and plant performance, and further illustrates the finer aspects of the integrated decision-making model as it was applied during the STP RI-IST project.

2.3.2.1 Qualitative Analysis of Limitations in the PRA

2.3.2.1.1 Truncated components

STP understands the significance of truncation limits set at inappropriately high levels. In the STP PSA, truncation limits are set at both the fault tree (i.e., system level) and event tree (i.e., plant level) levels. User-defined truncation thresholds are used for complex systems to facilitate the analysis relating to computer software limitations and run times. At the fault tree level, the user-defined threshold is referred to as the "cutset truncation." At the plant level, the user-defined threshold is referred to as the "sequence truncation."

Cutset truncation is the means of capturing enough cutsets from the fault tree to adequately describe the system for analysis purposes. The cutset truncation level is dependent upon the complexity of the system. For simple fault tree analysis, the cutset truncation does not require a truncation level to be established. That is, all cutsets for the fault tree are quantified and saved in the system analysis database. For large fault tree analysis with a cutset truncation limit set at zero, a portion of the captured cutset information will not significantly contribute to the overall failure probability of the system (i.e., this constitutes a large number of cutsets each with extremely low contributions). Clearly, a cutset truncation is sometimes desired for computer limitations like hard drive space and run time. In addition, the computer code imposes a cutset limit of approximately 11,000 cutsets for system level uncertainty calculation. In practical terms, the limit was set as low as possible while maintaining the uncertainty calculation cutset limit. In all cases, the analysis results in a cutset truncation limit which is less than or equal to 1E-12[22].

STP has set the "sequence truncation" limit to 1E-12. The sequence truncation limit represents the frequency at which individual accident sequences at the plant level are saved to the sequence database. The sequence database is used for computing the risk metrics (e.g., FV and RAW).

STP has set the sequence truncation for the On-Line Maintenance Program 1E-10. This truncation level is adequate for establishing the risk significance of plant configurations, while still allowing for a manageable quantification time to appropriately facilitate the program.

Finally, the truncation limits for sensitivity studies performed in support of risk-informed applications are the same as those used for the overall plant quantification.

2.3.2.1.2 Components Not Modeled In The PRA

A significant fraction of IST components or component functions are not modeled by the PRA (over 50% of the components considered for test interval extension). While it is likely that such components are not risk significant, the RI-IST Working Group evaluated each component and its associated design basis functions addressed by the IST program. Most components that are not in the PRA were found to be implicitly modeled by the study. That is, the PRA found that the components either were not required for the system to prevent severe accidents, were in systems that provided a highly redundant function, or performed functions that were unlikely to be required. The systematic review of these components by the RI-IST Working Group used quantitative and qualitative insights to determine whether component should be considered more or less risk significant and whether risk insights implied that compensatory actions should be considered. The narrative bases authored for each component group capture these insights. The bases reside in the RI-IST database.

The unmodeled components and functions were reviewed to determine their risk significance considering their potential roles in preventing core damage and/or large early release. If their function was considered to be important in this regard, these components and their associated functions were carefully documented and will be added to the PRA if appropriate via the PRA change process. Their equivalent importance was determined using insights gained from implementing the ranking methods discussed previously.

The first effort in assuring completeness in the ranking process was to compare PRA failure modes to IST component design basis function. To facilitate a general understanding of how the two types of functions compare, a detailed component and function level comparison was performed. This comparison essentially linked the PRA to the design basis, thereby allowing probabilistic and deterministic insights to be integrated in a traceable format.

There are two basic types of IST functions. The first maintains the integrity of fission product boundaries

[22] All system level truncation levels are less than 1E-12 and only one systems analysis is equal to 1E-12.

(generally, a closing function, often classified as a flow path boundary or isolation function), and the second ensures safety system operability (generally, an opening function, usually denoted as a "flow path" or sometimes as a "venting" function). A report in Attachment 3, "RI-IST Component Categorizations and Test Frequencies," lists IST functions (equivalent to IST tests, e.g., testing open or testing closed) for each component group, along with the RI-IST ranking categorization for that grouping.

The first type of IST safety functions ensures the integrity of the primary and secondary systems and provides containment isolation. Often these components are excluded because they mitigate highly unlikely scenarios. For example, the PRA often makes assumptions based on the low likelihood of certain scenarios that exclude from explicit models the possibility of IST valves failing to function. Examples of this include system pipe breaks occurring coincidentally with an accident, followed by an IST valve failure, or multiple failure of fail-safe valves.

The PRA also explicitly models most safety system operability functions. For example, most if not all components in the system flow path are modeled by the PRA. Exceptions to this, that is where system flow path is not modeled, include IST functions that are assumed to have low significance due to ample opportunity for operator action to recover, restore or establish an alternative. The following flow path functions assessed by the RI-IST Working Group to have low significance are:

1. Component Cooling Water (CCW) heat exchanger outlet flow path [CC07];
2. Air sampling flow path for the Containment Hydrogen Monitoring system [CM01];
3. Boric acid transfer (BAT) pump recirculation flow path [CV05];
4. Alternate boric acid makeup supply flow path [CV24 and CV41];
5. Charging pump discharge bypass flow path [CV32];
6. Essential Cooling Water (ECW) screen wash flow paths [EW03, EW09 and booster pumps];
7. Residual Heat Removal (RHR) heat exchanger return to hot leg [SI11]; and
8. Safety Injection (SI) accumulator vent flow paths [SI16, SI17, and SI26].

While in most cases IST functions for system flow path are modeled in the PRA, the PRA often does not explicitly model IST components that are intended to function to ensure the system flow path boundary is maintained. Such components are often implicitly modeled via PRA assumptions.

Given the development of this basic understanding of IST and PRA safety functions, a process was developed for evaluating components not explicitly modeled by the PRA. The process for evaluating such components depended heavily on two sources of information. One of the most important sources was the Risk Significance Basis Documents, which contain assumptions and system success criteria that indicate why some components or component functions are not required to mitigate certain accident scenarios.

The second source of information was the RI-IST Working Group knowledge of plant operations and design. Plant operations support and engineering support from the panel was used to rank a number of components, such as those associated the ECW screen wash and self-cleaning emergency backflush function [EW04, EW06, and EW07]. In this case for example, the frequency of planned use of the components, which depends upon an upstream dam failure event causing a need for the components in the system, was an important factor in the ranking. In other cases, the RI-IST Working Group served as an expedient source for understanding system operation and verifying the component failure modes that would have to occur and redundant components required to fail for the IST function to be needed. In these cases, documentation was provided which demonstrated that system failure modes were unlikely enough that components should be ranked low. The following table contains valve group discussions that illustrate the types of bases developed by the RI-IST Working Group for components that are not modeled.

VALVE GROUP	GROUP DESCRIPTION	SAFETY FUNCTION	BASIS FOR RANKING RI-IST LOW
CC05	CCW Common Suction Header Isolation MOVs - Trains A, B, and C	These valves must open to provide a return path from the Spent Fuel Pool Heat Exchangers, RCP thermal barrier heat exchangers, bearing lube oil coolers, and motor air coolers to the Train B pump if it is operating for accident conditions. In addition, these valves must close to isolate the return flow path from the Spent Fuel Pool Heat Exchangers, RCP thermal barrier heat exchangers, bearing lube oil coolers, and motor air coolers if the surge tank level is low or the pump has stopped.	This valve is normally open. Upon failure, this valve will remain in its failure position. The greatest risk is associated with the open function to provide CCW flow. Since these valves are normally open, this function is satisfied without operation of the valve. Reopening of the valve presupposes a previous need for closure [as described in the safety functions for this valve], meaning that a failure has already occurred in addition to the postulated failure of this valve to perform its function, an unlikely event. Moreover, there are three trains available to supply CCW flow, each with the same system configuration. Therefore, there is adequate redundancy in the capability of components to perform this safety function if called upon to do so.
CC10	CCW Supply (OCIV) to RHR Pump and Heat Exchanger - Trains A, B, and C	The valves must remain open to provide flow path for CCW through RHR pump seal cooler and RHR heat exchanger for accident conditions. These valves should close (remote manual) in response to a tube rupture in the RHR heat exchanger per UFSAR Section 6.2.4.2.1, Item 1.b and leak tight (CAT A) in accordance with UFSAR commitment (Section 6.2.6.3 and Figure 6.2.4-1, Sheet 35) to provide containment integrity.	This valve is a normally open motor operated valve. Since these valves are normally open, the opening function is satisfied without operation of the valve. Reopening of the valve presupposes a previous need for closure [as described in the safety functions for this valve], meaning that a failure has already occurred in addition to the postulated failure of this valve to perform its function, an unlikely event. A downstream check valve provides redundancy for the closing function. The MOV is designed with greater margin than needed to close against the higher pressure of the RHR system to isolate the system in the event of an RHR heat exchanger tube rupture. From an ISLOCA standpoint, the quantity of release from one tube failure is small. The likelihood of an event failing multiple tubes without failing the shell is extremely small. Additionally, the valve is in a physically closed system in which the piping has a higher design pressure than containment pressure and it is not connected to the reactor coolant pressure boundary. Finally, each train of RHR is functionally redundant, and only one train is required.

The evaluation was documented in the form of meeting minutes and in the form of component categorization narrative bases that reside in the RI-IST database. The RI-IST Working Group component bases identify the component group, the IST function(s), the RI-IST component categorization, compensatory actions (for potentially high components), and deterministic comments that often clarified the technical basis for the ranking.

2.3.2.2 High Risk Components Not in the IST Program

The IST ranking process identified many components for inclusion in the proposed RI-IST program. A handful of these components are non-safety-related pumps and valves, and are considered important to the operation of South Texas Project. However, none of these components have been designated by the RI-IST Working Group as RI-IST High. Nonetheless, RI-IST project team evaluated all of these components to determine the appropriate testing strategy. In the process, the team also identified for evaluation several safety-related components that are not considered to be traditional Code components, such as fans, dampers, and chillers. The PRA models these components. Their contribution to the plant's total risk spectrum suggests they warrant high risk rankings and an appropriate testing or performance monitoring strategy that ensures their continued reliability. Because of this recommendation, the RI-IST Working Group evaluated these components for inclusion in the RI-IST program. Each group of components considered for inclusion in the RI-IST program is described below, along with a strategy that should result in the continued or improved reliability of these key components.

The RI-IST Working Group reviewed the Main Steam Dump Valves and did not consider them to warrant the RI-IST High ranking. In its deliberations, the group noted that a current STP process has targeted these valves and developed an appropriate plan of action to improve their reliability. The plan of action includes the implementation of design changes to improve valve performance. If these modifications do not result in a reliability improvement, the RI-IST Working Group will consider these valves for inclusion in the RI-IST program.

Similarly, the RI-IST Working Group reviewed the Start-Up Feedwater Pumps, determining that they do not warrant a rank of RI-IST High. Functionally, these pumps may be available to provide water to the Steam Generators as a back up to four trains of Auxiliary Feedwater. The Auxiliary Feedwater trains are in the IST program and are considered to be adequate to provide the function. At this time, the Working Group has decided not to include the components in the program, but will revisit the decision during its periodic review of the program.

Finally, the RI-IST Working Group reviewed the Electrical Auxiliary Building Main Area Cooling system, which provides cooling to the area that includes the relay cabinets for the Solid State Protection System. PRA risk measures indicate that components in the system--such as fans, chillers, and dampers--are highly risk significant. It is not practicable to perform Code testing on these types of components. However, because of their importance, the Working Group evaluated the testing and maintenance being performed on the 33 fans, 6 chillers, and 21 dampers in the system. The RI-IST Working Group found that the components are tested frequently and adequately. The testing includes vibration measurements, operability verifications, and, in some cases, Technical Specification slave relay tests for fans. For dampers, scheduled maintenance activities assure the reliability of the equipment. A maintenance history review of these components identified no equipment failures in the last five years. Therefore, the RI-IST Working Group determined that additional testing provided by an IST program would not add value above that which is already provided by existing programmatic activities. However, as they are highly safety significant, the RI-IST project team will evaluate the existing monitoring process at the time of RI-IST updates (i.e., the RI-IST periodic review) to ensure the continued availability and operability of these components.

2.3.2.3 Completeness Issues (Sensitivity Studies)

Quantitative risk models have limitations associated with the structure of the models and the assumptions and the input data used. The limitations were compensated for by evaluating truncation limits, identifying IST components masked by the PRA, applying a conservative treatment of common cause failures, requiring an RI-IST Working Group to identify components with operational concerns, and performing selected sensitivity studies.

The risk ranking process described above used the FV and RAW importance measures. The values for these importance measures are calculated based on cutsets. The cumulative effects analysis described below also is based on cutsets. Cutsets are obtained by solving the model with a truncation limit. Experience has shown that setting the truncation limit arbitrarily low creates inefficiencies such that analysis costs quickly exceed the value of risk insights gained. This project evaluated the truncation limit used in the STP PRA and found it to be sufficient for both risk ranking and estimating cumulative effects.

The PRA model may "mask" certain components because they are associated with supercomponents (components which are internal to or mounted upon other components, e.g., pump internal check valves), human events, or initiating events but not explicitly identified. Masking occurs when the masking event (e.g., operator action) has an artificially high importance, potentially obscuring the importance of another component function. The components masked by the PRA model are typically small contributors to the overall probability of the event.

Risk ranking results can be strongly affected by the contribution of common cause failure. The approach taken in the project was to conservatively assume that a common cause event in the cutsets should have its entire risk significance assigned to all components represented by the event. This approach lead to the inclusion of a significant number of components in the more risk significant category which otherwise would have been considered less risk significant. The Expert Panel confirmed that the approach identified potentially important components.

Both risk ranking measures used are influenced by the reliability data assigned to the component. The STP PRA uses generic and plant-specific data since a previous study had indicated that STP component failure history on the whole is consistent with failure data reported to Nuclear Plant Reliability Data System (NPRDS). The Expert Panel considered whether or not plant-specific operational insights indicated component reliability problems that might affect the ranking of an individual component or small group of components. Components with operational concerns were considered more risk significant by the RI-IST Working Group.

Finally, the completeness of the models, assumptions and input data was tested by sensitivity studies. The sensitivity studies performed in support of STP's GQA Program considered most of the issues addressed by both the ASME Code Case and the NRC-approved RI-IST projects (i.e., TXU's Comanche Peak and SCE's San Onofre Nuclear Generating Station).

In the analysis phase of the GQA risk-informed application, STP performed a variety of sensitivity studies to provide additional assurance that important SSCs are not inappropriately categorized because of PRA modeling limitations and uncertainties. Toward this end, STP performed the following bounding values and analyses:

- Removal of all CCFs,
- Studying the potential degradation of availability of nominally identical components used in several systems, evaluated by assessing the impact of a common increase in unavailability,
- Setting equipment planned to be out of service during each of the plant's scheduled maintenance states to an unavailable state,
- Removal of all operator recovery actions, and
- Studying the effect of a possible over-estimate of induced steam generator tube rupture (SGTR) overshadowing other LERF considerations.

For CCFs, the sensitivity study considered the influence of CCF on component categorization. First, because CCF dominates risk, its contribution can mask individual component failure modes. No masking was found. Second, the results of the CCF analysis can be sensitive to the selection of CCF groups. In this case, it was assumed that every IST component group was a logical common cause group. This assumption was deemed reasonable because the IST component grouping methodology considers the most important factors related to CCF, namely component design and service condition. The CCF study provided further evidence of both the quality of the STP PRA and the robustness of the categorization method. When the potential degradation of availability of nominally identical components used in several systems was evaluated, the results indicated no change to the component categorization.

For maintenance unavailabilities and removal of operator recovery actions, the issue was again the possibility of masking. The sensitivity results indicated no potential for masking.

Finally, induced steam generator tube rupture contributes greatly to LERF in the STP PSA. To determine the effect of SGTR event assumptions on risk ranking, STP performed a sensitivity study that reduced the assumed probability of an induced SGTR by one half. The sensitivity results indicated no potential for masking due to uncertainties associated with this postulated event.

In conclusion, the sensitivity studies performed were comprehensive and addressed the intent, if not the form, of the sensitivity studies recommended by the ASME OMN-3 Code Case addressing the component categorization process. Moreover, after assessing the bounding values and analyses used to support the categorization process, the NRC has deemed the sensitivity studies to be adequate for the purpose of assigning components "(in relation to their importance to the CDF and LERF risk metrics) into broad safety-significance categories for consideration by the WG and Expert Panel."[6]

2.3.2.4 Integration with Other STP Risk-Informed Applications

A linkage exists between the categorization of RI-IST components and the categorization of these same components in GQA, Maintenance Rule, and other plant risk-informed programs. In general, the risk rankings for these applications should be similar because the PRA is used for all component categorization efforts at STP. However, IST tests only for active failure modes. Therefore, the PRA risk measures used in the RI-IST component categorization effort include only active failure modes. As expected, this circumstance results in occasional differences in component categorizations across plant programs because other programs may consider additional failure modes, such as passive failure modes. Moreover, programmatic efforts may place slightly different emphases on factors contributing to the component categorization process, or some may consider attributes that do not logically lend themselves to inclusion in other programs. For instance, the GQA program incorporates elements of organizational performance (i.e., plant organizational effectiveness versus maintenance effectiveness) that is not an element of either the Maintenance Rule or IST. Nevertheless, in its deliberations, both the RI-IST Working Group and the plant Expert Panel made every effort to remain consistent with component categorizations associated with other programmatic activities, and to understand why differences in the component rankings should exist when the case arose.

In addition to risk-informing programmatic activities, STP has recently requested to exclude some components from the scope of special treatment required by regulations. That submittal includes a request for exempting low-ranked components from IST. At this time, the NRC has issued a draft safety evaluation report (SER)[7] that offers preliminary acceptance of exempting GQA Low components from the scope of IST.

However, this submittal focuses on delineating an RI-IST program that complies with guidance outlined in RG 1.175 (i.e., no scope changes). Upon issuance of regulatory acceptance of this relief request, STP plans to implement the RI-IST program evaluated in this document and outlined in Attachments 2 and 3. When the NRC issues its final acceptance of the exemption request, STP will, at that time, implement the program as outlined in the exemption request. That is, those components ranked GQA Low and not risk significant (NRS) will not be included in the scope of the RI-IST. However, the remaining components will receive the programmatic treatment described in Attachment 2. As discussed in Section 2.3.3, based on the nature of the risk changes--namely that postulated risk increases are very small; the direct and indirect safety benefits, which are widespread, possibly substantial and on their own should reduce uncertainty; and then finally on the consistent level of conservatism and justification provided for assumptions used in the calculations -- the conclusion is that implementation of the RI-IST program will be either risk beneficial, or at most risk neutral.

2.3.3 Use of the PRA to Evaluate Effects of Proposed Changes on Risk

The final component categorization does not necessarily guarantee that acceptable levels of risk will result in the RI-IST program. Changes to many components simultaneously may cause unintended increases in risk, despite meeting the conservative risk ranking measures selected. Therefore, an analysis was performed to determine the effect of all RI-IST program changes on total plant risk. This analysis is intended to:

- Model the impact of various RI-IST program changes (i.e., interval extensions and compensatory measures),
- Evaluate the resulting effect on total plant risk (i.e., total core damage frequency and total large early release frequency), and then
- Compare the effect of RI-IST program changes to acceptance criteria in RG 1.174.

The impact of program changes was modeled considering available information on how changes in test intervals will change component performance. Uncertainty in this input information, together with the complexity required to model such an approach, dictated the use of a number of assumptions and judgements.

The effect on total plant risk was evaluated using a full re-quantification of the STP RISKMAN® model. The model includes quantitative estimates for external events. This calculation was complemented with judgement for items not directly represented by the PRA.

Finally, the discussion shows how the STP RI-IST program satisfies acceptance criteria from RG 1.174 and RG 1.175.

The following sections describe the assumptions, calculations, and judgements made.

2.3.3.1 Modeling the Impact of Changes in the IST Program

An analysis was performed to determine the potential risk impact of increasing in-service testing intervals simultaneously on all less risk significant components. Consideration was given to available information on how changes in test intervals will change component failure probabilities, common cause failure probabilities, and initiating event frequencies.

Component Failure Probabilities. Uncertainty in the available information, together with the complexity required to model such an approach, dictated the use of a number of assumptions for calculating changes in component failure probabilities:

- It is assumed that any increase in test intervals would simultaneously impact the reliability of all IST components in the RI-IST Medium and RI-IST Low categories.
- Consistent with the PRA techniques, the component failure on demand, Q_D, is assumed to be:

  $$Q_D = f_s{}^*Q_S + (1-f_s)^*(\lambda T)/2$$

  where,

f_s = fraction of total failure rate assigned to demand failures

Q_S= the component failure due to change in state (shock),

λ = the component standby failure rate per hour, and,

T = the interval between tests (hours) that verify operability of the component.

- The component failure on demand is assumed to increase by the same factor as the increase in the test interval (i.e., linearly increases with the time between tests). This is accomplished in the RISKMAN models by setting the fraction f_s to 0. For example, a change in the test interval from quarterly to semi-annually is assumed to increase Q_D by a factor of two.
- Decrease in wearout due to less frequent testing is assumed to be negligible although frequent testing has been seen to cause components to be less available due to wearout.
- It is conservatively assumed that all IST tests are fully effective in finding the causes of component unavailability.

The following discussion reviews the potentially non-conservative assumptions used in modeling the effects of RI-IST program changes and justifies why they are not considered significant. Those assumptions are:

- Fully effective compensatory measures
- Constant failure rate, namely no impact from aging

The calculation assumes that compensatory measures are fully effective or otherwise equivalent to the IST. The compensatory measure that is most relevant is the slave relay test for MOVs and AOVs. The assumption presumes that the fault finding capability of the relay test is equivalent to the IST. This assumption is consistent with both traditional and probabilistic techniques.

Regarding traditional considerations, the MOV or AOV must function for the relay to pass its Technical Specification surveillance. The compensatory measure consequently determines whether the MOV or AOV functionally fails. Regarding probabilistic factors, the measure is essentially equivalent to a surveillance test. In PRAs, a surveillance test interval would typically be credited as the test interval in a failure probability calculation. (In the case of the slave relay test, the compensatory measure was credited at its Technical Specification prescribed six-month interval for applicable components. Hence, the failure probability for an RI-IST Medium component with this compensatory measure was increased by a factor of two, a value equivalent to a test interval increase from 3 months to 6 months.)

While the assumption of equivalent fault finding capability is justified, many compensatory measures were not credited in calculations reported in the next section:

- Those required by the STP RI-IST program for RI-IST Rank Medium components
- Normal system evolutions
- Equipment rotations for run-time equalization

Consequently, the treatment of compensatory measures is also conservative.

The constant failure rate assumption considers no impact from aging. In a critique of the ASME approach to risk-informed IST[23], Dr. William Vesely states that the component importance should be determined using failure probabilities (unavailabilities) that depend on the age of the plant, even if constant failure rates are assumed. He further states that large variations in the failure probabilities can occur when plants are categorized according to their age.

In PRAs, the component failure probability is usually assumed to be constant based on the assumption that the changes in component failure probabilities follow the bath-tub curve. That is, the failure probabilities are constant for the majority of the plant life before they start deteriorating due to aging. The STP RI-IST program considered the effect of aging. However, no major evaluation was judged to be necessary for the following three reasons.

First, one of the major elements of the RI-IST program is performance monitoring. If any changes to the IST program lead to a gradual equipment degradation and a resulting performance problem, the problem will be quickly identified through root cause analysis and the corrective action program. The RI-IST program requires periodic updates and necessary modifications to correct any performance problems due to either aging or any other plant-specific operating practices. Therefore, the program itself will identify and correct potential age-related performance degradation.

Second, the STP RI-IST program recommends that the test intervals of the IST components in the low risk significance category be extended to every 18 months to 6 years depending on IST group size. Consequently, the monitoring program will yield component performance data for many different test intervals. The understanding of component performance under the effect of aging should actually improve under the RI-IST program.

Third, a study was done by Dr. Vesely to show the unavailability changes for check valves versus IST intervals for various valve aging rates[24]. The results collectively showed that, up to approximately a 10-year test interval, the unavailabilities stayed at or below the component unavailability at the test interval of once per quarter. This study seems to support the test intervals of 2 to 8 years for low safety significant check valves.

Since the tests on the components will be staggered, and since component performance will be monitored (in some cases with enhanced test methods), corrective action can be taken to effectively remove or correct for any degradation mechanisms such as aging. Hence, the assumption of constant failure rates is justified.

Uncertainty in aging effects from extended test intervals is offset somewhat by the conservative assumption that there is no impact from testing-induced wearout effects. In performing this study, we did

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23 Memorandum from Dr. William E. Vesely of SAIC to Mr. Mark Cunningham of NRC, "Reservations with ASME Risk-based Inservice Inspection and Testing," April 17· 1996.

24 NUREG/CR-6508, "Component Unavailability versus Inservice Test (IST) Interval: Evaluations of Component Aging Effects with Applications to Check Valves," developed by Oak Ridge National Laboratory for the NRC's Division of Engineering Technology Office of Nuclear Regulatory Research, July 1997.

not comprehensively review and evaluate existing studies on wearout or test-induced unavailability. However, studies lend credence to the possibility of negative influences of testing on total component failure probability[25]. Conclusions of these studies suggest that "too frequent testing" is a stronger negative influence on component failure probabilities than "too infrequent testing". These observations imply that it is conservative to extend intervals when uncertainty exists.

IST may be particularly sensitive to this effect because of its focus on component performance degradations. One of the important contributors to negative impacts on unavailability from testing occurs when a test or preventative maintenance (PM) finds a degradation which is not a functional failure, but which causes the component to be removed from service for corrective maintenance. In other words, unavailability in this case is assured because the component is "prematurely" removed from service.

Moreover, for much of the factor increase in test intervals from the current test interval, data on "aging" does exist. Since many ISTs are now done on a refueling cycle basis, the RI-IST program benefits from this existing test experience when extending test intervals from 3 months to 2 years. The paucity of data on aging relates to the 2-year to 8-year portion of the change.

In the case of 2 to 8-year interval changes, many older plants have valves in power piping code systems that are identical to or at least similar to Code Class 3 valves that are subject to IST. To our knowledge, data that compares the reliability of these valves have not been published. However, indications from plant-to-plant variability in generic valve failure data apparently contradict our conservative assumption of large factor increases in some component failure probabilities. A valve initially assumed to fail at 3E-03/demand on a quarterly test interval is assumed in our calculations to have a 0.1/demand failure rate if the RI-IST program specifies an 8-year staggered test and no compensatory measure. However, plant-to-plant variability in generic data indicates that, assuming an error factor of 10, an initial 3E-03 has a 95% upper bound of 0.01, and a 99% upper bound of 0.03. Typically, IST components exhibit error factors less than 10, so the upper bound is much closer to the mean value. Consequently, present generic data do not support valve failure probabilities as large as those assumed in our calculations.

While PRA methods guidance is typically silent on the topic of infrequently tested components, what guidance does exist suggests that our calculations are conservative. For example, the IREP PRA Guidance documents suggest using the 95% upper bound value for an infrequently tested component.

In summary, the two potentially non-conservative assumptions –those associated with fully effective compensatory measures and a constant component failure rate-- are justified by the arguments above. Potential non-conservatisms are further compensated for by programmatic elements in the RI-IST program, such as staggered testing and performance monitoring. Therefore, the [(λT)/2] model can be considered adequate for application to component failure probabilities.

[25] E.V. Lofgren, et al., "Nuclear Power Plants Standby and Demand Stress Component Failure Modes: Methodology, Database, and Risk Implications," prepared by SAIC for US NRC Divisions of Systems Research Probabilistic Risk Analysis Branch, February 1992.

Common Cause Failures. As discussed above, the common cause failure probabilities can also increase with IST interval changes. The most conservative time between testing was assumed for the CCF value estimate for the factor increase in failure rate. The following examples illustrate how common cause values were increased to model IST interval increases:

1. A CCF group with valves originally tested on a quarterly basis, now tested once every 6 years with one valve in the group of four tested every 2 years (also referred to as 2-year staggered testing) - the associated common cause failure on demand probability is effectively increased by a factor of 8 to reflect the 2-year interval using the basic event probabilities described previously.

2. A CCF group including valves whose interval was not extended and valves whose interval was extended – the CCF probability was generally not changed. Since some of the valves are still tested on the same test schedule, the common cause group test interval is generally unaffected. However, the test schedule was reviewed to ensure the time between tests for components in the group remained unchanged.

3. A CCF group including valves whose RI-IST intervals are different (e.g., one tested every 2 years and one tested every 6 years), was based upon the shortest time between tests (in this case, 2 years).

4. A CCF group whose group interval remained the same, but the component tests were staggered, did not have the common cause changed. Consider, for example, a valve group that was originally tested every 2 years during shutdown, i.e., each valve in the group tested every 2 years. If the RI-IST program incorporated staggered testing such that one of the valves was tested every 2 years, the common cause failure probability was not increased.

Accordingly, the modeling of CCF changes due to IST program changes reflects the significant risk benefit that can result from implementing the staggered testing philosophy suggested by RG 1.175.

Initiating Events. The RI-IST program is not expected to have a significant effect on the initiating events included in the South Texas PRA. Two systems which contain components subject to IST are modeled as Support System initiating events.These systems, essential cooling water system (EW) and the component cooling water system (CC), contain components which are ranked High and Medium respectively. These two systems are rotated weekly for maintenance activities and as a result, each train is challenged. The EW and CC system pumps perform their required safety function (i.e. start on demand) and valves in these systems are repositioned. The PRA takes into account these demands on system performance therefore, no changes in test frequency or method modeled by the PRA are proposed for these systems.

Conclusion. Modeling the effects of changes in the RI-IST program requires changes to individual component failure probabilities, which in turn affect common cause failure probabilities and initiating event frequencies. The $[(\lambda T)/2]$ model can be considered adequate for these applications because conservatisms and programmatic elements such as staggered testing and performance monitoring compensate for potential non-conservatisms in the model.

2.3.3.2 Evaluating the Change in CDF and LERF

Evaluating the change in CDF and LERF was done in a two-step process. First, using certain assumptions, a comprehensive bounding calculation was performed using the STP PRA software. Second, the evaluation included an estimate of the impact of other safety benefits, including those that result both directly and indirectly from the RI-IST program. The following describes the STP PRA scope and the bounding calculations. This section then describes the other safety benefits and reaches the conclusion that the RI-IST program will result in safety neutrality.

2.3.3.2.1. Bounding Estimate of the Change in CDF and LERF

STP PRA Scope. The current STP PRA, documented as STP_1997, includes all external events and is a complete level 2 analysis of core damage frequency and large early release frequency of the South Texas Project Electric Generating Station. Total plant risk has been evaluated in a comprehensive manner. For this reason, the impact of IST program changes on CDF and LERF were calculated directly without making approximations for most risk sources.

It is worthy of note that the total plant risk is at a favorable level compared to the acceptance criteria in RG 1.174. The total change in plant CDF is 1E-7 per year and total change in plant LERF is 1E-9 per year. Both changes in CDF and LERF are well below their respective RG 1.174 acceptance criteria of 1E-6 per year and 1E-7 per year, respectively.

Bounding Calculations. The calculations indicate that, using bounding assumptions, the CDF and LERF risk increases are small (0.9% and 0.2%, respectively).

Average Maintenance Bounding Analysis

RISK METRIC AND MAGNITUDE	CDF CHANGES	CDF FRACTIONAL CHANGE (%)	LERF CHANGES	LERF FRACTIONAL CHANGE (%)
Increases due to interval extensions	1.E-07	0.9	1.0E-09	0.2

The impact of the remaining safety benefits were estimated, rather than calculated. Their impact is discussed in the next section. As discussed in the previous section, only those regulatory driven compensatory measures (e.g., slave relay tests) are credited. The benefit of other compensatory measures has not been estimated. That calculation was deemed unnecessary given the very small increase in CDF and LERF.

2.3.3.2.2. Estimate of the Change in Risk Due to Direct and Indirect Safety Benefits

The bounding risk estimates conservatively do not consider many of the safety benefits from the proposed program. This is significant and necessary for the calculation because:

- Some uncertainties exist in the impact the safety benefits would have on model parameters,
- Some of the benefits are qualitative in nature and are very difficult to quantify, and
- Some aspects of program implementation that affect the safety benefits have not yet been finalized.

The following describes the important safety benefits and estimates their significance.

The STP RI-IST program will provide the following safety benefits as a direct result of IST programmatic changes:

- Reliability improvements for RI-IST High components in the IST program:
  1. Reduction in exposure to potential system re-alignment errors
  2. Improved performance resulting from improving the quantity and quality of plant personnel time devoted to RI-IST High components
- Reliability improvements for RI-IST Medium components (i.e., the LHSI pumps) in the IST program.

The STP RI-IST program will also provide indirect safety benefits such as:

- Reduction in human errors due to a reduction in operator burden
- Improved system failure probabilities upon demand due to fewer off-normal operational line-ups
- Other safety impacts related to improvement in safety culture:
  1. Improved understanding of component level importance
  2. Monitoring of CCF components
  3. Operator awareness of important PRA failure modes for IST components

The following estimates the potential risk impact of direct safety benefits that are not accounted for in the PRA calculation for the reasons mentioned above. Possible impacts from the indirect safety benefits are subsequently noted.

Combining the bounding estimate using the STP PRA calculation tool with the more limited quantification of direct safety benefits indicates that total plant CDF and LERF could potentially be reduced as a result of changes to be implemented in the RI-IST program. The estimated reductions in CDFand LERF are on the order of 5%.

Direct Safety Benefits. Possibly the most important effect of the proposed RI-IST program will likely be the reliability improvements for RI-IST High components in the IST program, as it is expected that increased attention and reduced manipulation of these components will improve reliability and decrease unavailability due to human errors. Also, with fewer tests, system line-up/realignment errors are less likely. For example, it is estimated that since the total pump unavailability (not including latent human error) is in the range of 5E-3, performance improvements might range from a few percent to tens of percent. The system realignment with the most impact on train unavailability due to latent human error is often the pump alignment. Pump alignment typically remains unchanged when the pump is categorized as an RI-IST High component (systems AF, ECW, and HHSI). Hence, the improvement to a typical RI-IST High component due to this safety benefit might be less than one percent.

Improved safety margins should result by focusing resources on high risk components and reducing the testing frequency on low risk components. One can make the assumption that there is a limited amount of Operations and Maintenance (O&M) resources available for programs such as IST. Then, any reduction in the IST program activities assures that the O&M resources that are available are spent in an increased fraction on the RI-IST High components and not diluted by work activities that have an insignificant impact on risk. In this sense, the IST O&M resources are focused on the RI-IST High components. For example, the IST engineer and system engineers will have more time available to analyze trends in component and system performance data. Because more types of data will be available to trend or compare (e.g., components with varying IST intervals, or possibly components added to the IST program in the future), this increased time may further develop into a better understanding of the factors which influence component performance and reliability. The former is discussed in Section 2.3.2 under safety margins.

The impact of this improvement in safety margin is hard to measure, but generic data on plant variability indicates the best performing high risk components could easily be better by a factor of three or more than poorly performing high risk components (in terms of individual component contributions). It seems reasonable to assume that a few percent increase in RI-IST High components is extremely plausible in the near term, with possibly additional increases in the longer term.

Regarding component reliability improvements due to testing enhancements to be proposed by ASME, there is some hope that these improvements could be significant. ASME has devoted considerable research to the causes of pump failures in particular. The NRC has sponsored research through Oak Ridge National Laboratory (ORNL) that is attempting to measure the effectiveness of certain test methods, including the comprehensive pump test. It does not seem unreasonable to assume that a few percent increase in component reliability could result, especially for pumps.

For example, a revised testing strategy for the LHSI pumps will be an important safety effect due to the potential CDF improvement value of these components. Currently, these components are tested in a mini-flow configuration, which can be potentially damaging to components on the line over a sustained period of time (i.e., with regard to vibration tests). STP proposes to replace the quarterly mini-flow test with a full flow test performed during refueling outages. This test is generally considered to be much more effective at detecting degradation that could potentially lead to failure of the component to perform its safety function than the current test. Furthermore, as the full flow test requires that components perform their functions at design or near design conditions (i.e., the optimum testing environment), this test is generally considered by industry experts to be less damaging to active components. Were inclusion of the full flow test to lead to better knowledge of the capability of the pump, one could conservatively postulate an improvement in the CDF resulting from this enhanced test strategy.

The impact of inservice testing on component reliability is not well known. However, it might be logical to assume that the amount of improved reliability due to testing enhancements would be similar to the factor of degradation assumed for components for which test intervals are increased. Comparing FV measures is equivalent to this assumption. Since the summed FV of the LHSI pumps (0.4% of CDF) is on the same order of magnitude as the "equivalent FV" for all RI-IST Medium and RI-IST Low components whose test interval has increased, it is possible that test improvements in the RI-IST program from the LHSI pumps alone could ensure the program is at least safety neutral, or very close to safety neutral.

It is also worth noting that changes to IST intervals and the scope of components included will provide more information with which to identify the most effective testing methods. Therefore, the STP implementation of RI-IST may eventually provide further improvements to ASME's efforts.

Indirect Safety Benefits. The following indirect safety benefits are not accompanied by estimates of quantitative improvements. Taken as a whole, however, they could be substantial since they deal with plant-wide improvements in safety.

Perhaps the most difficult safety benefit to measure might be the amount of reduction in human errors that might result from a reduction in operator burden. STP has noted that senior reactor operators (SROs) and reactor operators (ROs) will spend fewer man-hours performing system line-ups for testing and realignments after testing and performing work package reviews. Since human errors are involved in almost every important cutset in a PRA, an improvement in average operator failure probabilities may cause a similar reduction in CDF and LERF.

STP also expects that improved system failure probabilities upon demand could result due to fewer off-normal system alignments. PRAs generally assume normal system alignments. Traditional safety programs often make the same assumption. Such conditions (i.e., systems not in their normal alignment) have the potential to cause unanticipated problems, mostly due to less experience with them. Generally a normal alignment will require fewer components to actuate. In particular, a normal alignment will require fewer "less frequently functioning" valves to operate, e.g., system boundary isolation valves, manual valves, and test return line valves. Also, operators will need to operate manual valves less frequently in demand situations, if the time in off-normal conditions is reduced.

Another important indirect safety benefit that will result from implementation of RI-IST is the improvement in safety culture that can result from a site-wide improvement in understanding of the important contributors to risk, including:

- Improved understanding of component level importance,
- Monitoring of CCF components, and
- Operator awareness of important failure modes in IST components.

It could be argued that such improvements are already occurring as a result of increased awareness of the PRA, implementation of the Maintenance Rule, and use of risk management during outages and on-line maintenance activities. However, the improved understanding of component level importance and the increased emphasis on monitoring for common cause failure could result in important safety improvements. The more such improvements are integrated into the safety culture by changing common plant programs such as IST, the more these benefits will be realized.

Summary. In conclusion, implementation of the STP RI-IST program will result in at least risk neutrality, if not a net safety benefit. Further, both the direct and indirect benefits are potentially larger and more widespread than the limited risk changes indicated by the bounding analysis.

2.3.3.3 Comparison with Acceptance Guidelines

The RG 1.174 acceptance criteria depend on the total risk estimate and the estimated risk change. Because both CDF and LERF are well below the RG 1.174 acceptance criteria, a risk increase is permitted. However, as the discussion below indicates, the RI-IST program is safety neutral.

Using judgement to estimate safety benefits for the above-mentioned factors, the following table estimates the change in risk associated with the proposed program changes:

PROGRAM CHANGE	**CHANGE IN MODEL ELEMENT**	**ESTIMATED APPLICABLE FRACTION OF CUTSETS**	**TOTAL SAFETY IMPROVEMENT ASSUMED**
enhanced testing for selected components (e.g., LHSI pumps)	Improvement in reliability is likely the same as degradation in low risk components	4E-03	4E-03
reduction in system re-alignment errors	< 1%	8E-01*	5E-03
improved performance resulting from improving the quantity and quality of plant personnel time devoted to RI-IST High components	few %	8E-01*	2E-02
component reliability improvements due to testing enhancements to be proposed by ASME	few %	8E-01*	2E-02
reduction in human errors due to a reduction in operator burden	not estimated	~1.0**	Not estimated
improved system failure probabilities upon demand due to fewer off-normal operational line-ups	not estimated	8E-01*	Not estimated
other safety impacts related to improvement in safety culture	not estimated	~1.0**	Not estimated
Total Program Improvement			> 5E-02

*estimated

**assumes the issue is applicable to essentially all cutsets

The table indicates that it is reasonable to estimate that about a 5% improvement in CDF and LERF will result from the proposed program changes (since the bounding estimate yielded a less than 1% increase for CDF and LERF).

While this evaluation did not include a comprehensive uncertainty analysis such as that suggested by RG 1.174, the results of the assessment have been consistent. This conclusion is based on the nature of the risk changes, namely that postulated risk increases are very small; the indirect safety benefits, which are widespread, possibly substantial and on their own should reduce uncertainty; and then finally on the consistent level of conservatism and justification provided for assumptions used in the calculations. The

STP PRA has been demonstrated to be of a quality consistent with the requirements for this application and has been reviewed by the NRC for other risk informed plant applications. Finally, the program of monitoring, feedback, and corrective action is an important factor in addressing uncertainties related to the impact of degradation mechanisms and aging effects.

Consequently, the results show that the STP RI-IST program satisfies the acceptance criteria of Regulatory Guide 1.174 and that when combined with the tangible, qualitative risk benefits of enhanced testing of selected components and reduced testing of low risk components, the overall impact of the STP RI-IST is either risk beneficial, or at the very least, risk neutral.

2.4 INTEGRATED DECISION-MAKING PROCESS (IDP)

The role of the STP's IDP was crucial in ensuring that the results presented in this submittal are comprehensive. At STP, the RI-IST integrated decision-making process requires the participation of two member groups:

1. A plant Expert Panel, which is a multi-disciplinary group of individuals whose purpose is to guide the implementation of Comprehensive Risk Management activities at STP, and
2. An RI-IST Working Group, which is a multi-disciplinary group of individuals who provide risk-informed, performance-based recommendations to the plant Expert Panel.

The RI-IST Working Group members are senior level personnel whose membership has been endorsed by the Expert Panel. The RI-IST Working Group consisted of members with expertise in the areas of

- Power plant operations*,
- Plant maintenance*,
- PRA and nuclear safety analysis*,
- Systems engineering,
- Design basis engineering*,
- Safety analysis (Chapter 15)*,
- Quality assurance,
- Licensing, and
- Inservice testing (including ASME B&PV Code Section XI and ASME Code Cases)*.

* denotes voting members. Five voting members are required for quorum.

All the members of the RI-IST Working Group have at least ten years experience in nuclear power.

The IDP effort entailed RI-IST Working Group review and validation of the PRA risk measure, a process that ensured an integrated effort through active technology transfer. In addition to considering the basis for the PRA risk measure for modeled components, the RI-IST Working Group qualitatively assessed the following for each component group:

- The degree to which component failure leads to an increase in the frequency of initiating events,

- The degree to which component failure leads to the failure of another safety system,
- The degree to which component failure causes a transient,
- The role of the component in the plant EOPs or SAMGs, and
- The role of the component in plant shutdown.

As part of the process, the RI-IST Working Group authored a narrative basis to support the final RI-IST categorization of each component group.

Subsequent to Working Group initial RI-IST categorization of components, the STP plant Expert Panel considered and ultimately validated the results of all Working Group activities and studies performed by the IST project members. The Expert Panel consisted of members with expertise in the areas of power plant operations, plant maintenance, PRA and nuclear safety analysis, design engineering, and quality assurance. The Expert Panel served as the central point of decision-making for major technical issues and offered guidance to risk-informed IST project members in performing their work.

It was concluded that the strength of this risk-informed IST program and the integrity of its results lie both in the comprehensiveness of the methodology and in the work of both the RI-IST Working Group and the plant Expert Panel.

RI-IST Working Group Charter

To prepare for the Expert Panel review, the RI-IST project team used a process similar to that employed by TXU and SCE during their RI-IST projects. The PRA risk categories were displayed on simplified P&IDs to help illustrate for the RI-IST Working Group the roles redundancy and reliability play in risk categorization. Additionally, design basis functions were compared to PRA failure modes to clearly establish the relationship between PRA and the design basis.

The RI-IST Working Group used plant knowledge, operating experience, and engineering judgment to perform the following tasks:

- Verify component functional failure modes
- Establish risk-informed categorizations for components not modeled in the PRA
- Assess or provide qualitative deterministic criteria
- Consider and/or provide insight concerning the component performance history. Specific attention was afforded to areas of poor or declining performance.
- Address all significant safety and operational concerns
- Validate component categorizations
- Resolve questions relative to PRA model completeness
- Resolve all questions raised during the review process

The RI-IST Working Group considered the following factors in addition to the combination of risk significance and deterministic insights discussed above:

- Important design basis functions not reflected in the risk categorizations

- Impact of PRA scope limitations, assumptions, and model simplifications, such as exclusion of shutdown states
- Importance of release states less severe than large early releases that are not explicitly reflected in the risk categorization scheme

The RI-IST Working Group also considered as part of their evaluation the uncertainties caused by:

- PRA model assumptions
- Common cause or common mode failure rates
- Treatment of support systems
- Level of definition of cutsets and cutset truncation
- Model assumptions relative to repair and restoration of failed equipment
- Human error rates used in the PRA
- Limitations in the meaning of importance measures

Based on the process outlined above, the Working Group made a qualitative assessment of the RI-IST importance categories that were developed for the components using the PRA results and deterministic insights, plant-specific history, engineering judgements, and probabilistic risk analysis insights. The Working Group reviewed the PRA component risk rankings, compared the PRA and IST functions to ensure consistency with plant design, and analyzed applicable deterministic information in its effort to resolve the final safety significance categorizations for all the IST components scrutinized.

Documented recommendations developed by the RI-IST Working Group and forwarded to the Expert Panel included:

- RI-IST categorization and proposed test interval (i.e., no extension, extension with compensatory measures, or extension without compensatory measures)
- The bases for making those recommendations (i.e., including PRA inputs, performance analysis results, details regarding any other deterministic inputs)
- Identification of components not within the scope of the PRA, including components supporting balance of plant operations, mode transition and shutdown operations

The Expert Panel approved the final IST categorization (and, hence, the test interval for which the component is eligible) and proposed changes to the IST test program by reviewing and concurring with the recommendations of the RI-IST Working Group.

2.4.1 Corrective Maintenance Evaluation

A significant deterministic input to the decision-making process proved to be the component corrective maintenance evaluation performed by the RI-IST project team members. To facilitate the evaluation, the RI-IST project team took advantage of reports produced by STP's Operating Experience Group (OEG), which compiles and analyzes performance of plant equipment and activities. Data for the reports is compiled from various sources, including the Corrective Action Program (CAP) database and an equipment history database. The data is analyzed for performance trend changes. Any components with a poor performance or whose performance is on a declining trend are highlighted for evaluation.

In addition to analyzing OEG reports, the RI-IST project team performed an independent component maintenance history review, spanning several years (encompassing at the very least the period of time between 1/95 and 5/00). Conclusions about component performance were based on the tested IST function(s) for a given component. That is, if an event involved a failure of a valve to open, but IST tests the reliability of the valve to close (i.e., not to open), then the event was not considered to be an IST failure.

Example of a Performance History Review for the Auxiliary Feedwater System

To support the GQA Program risk-informed effort, the OEG conducted a review of the Auxiliary Feedwater (AF) system and subsystem events captured in NPRDS, the STP Corrective Action Program (CAP) database, and the AF Reliability History. The conclusions of their review are as follows:

- The Operating Experience Group reviewed the reliability history for the Auxiliary Feedwater System from January 1, 1995 through October 31, 1998. They identified five failures, two of which did not involve the valid equipment failure of Auxiliary Feedwater components. The other three failures consisted of electrical failures associated with motor-operated valves. These failures shared no commonality.
- The Condition Report (CR) database documents 430 documented conditions between January 1, 1995 and December 31, 1998 for the AF system. Of these 430 Condition Reports, the OEG determined that 160 involved valid component failures. The OEG identified no commonalities between these failures, with the exception of 22 that were directly attributed to human performance errors.
- The Institute of Nuclear Plant Operations NPRDS was evaluated for failures meeting the NPRDS reporting criteria. Of the 154 component failures documented between January 1, 1995, and December 31, 1997, the South Texas Project did not incur any component failures that met the reporting criteria.

Therefore, based on this review, the OEG agrees that the components in the system have adequate performance histories and are eligible for downgraded quality assurance activities.

To verify the results of the OEG review for the RI-IST Program, the RI-IST project team performed a corrective maintenance history review on AF pumps and valves within the scope of the IST Program. A search identified 329 preventive and corrective maintenance activities performed since January 1, 1995. Of these activities, the team identified five failures, with four of these failures resulting in the loss of a safety function tested by the IST Program. The failures are listed in the following table.

COMPONENT	FAILURE	CAUSE
C1AFMOV0085	Failed to open	Motor burned up, cause unknown
D2AFMOV0019	Failed to open	Oil film on electrical contacts
D1AFFV7526	Failed to open	Limit switch was not closed, adjusted switch finger to make contact
D2AFMOV0514	Closed, but did not re-latch	Failure could not be duplicated, cleaned torque switch contacts and bypass contacts.

The paucity of events in the above table indicates that failures have been infrequent for IST components in the Auxiliary Feedwater system. The identified failure cause of these events is different for each case, indicating that a common deficiency or inherent flaw in the design of the components does not exist.

Based on the above information, the Auxiliary Feedwater system components at South Texas Project have performed reliably and can be tested at an extended frequency as determined by their RI-IST safety significance.

Poor Performers

Once the corrective maintenance history had been fully reviewed for a component, a summary of failure events or particularly eventful corrective maintenance histories was reported to the RI-IST Working Group for their consideration during the risk categorization process. This was useful in facilitating the determination of contentious performers (i.e., those components for which the RI-IST Low categorization merits assigning either a compensatory measure, retaining the current test interval, or changing the ranking to RI-IST High). The RI-IST Working Group changed the rankings of only one component group, MS03, the power-operated relief valves, to RI-IST High as a result of this maintenance history review process.

In addition, the RI-IST Working Group determined that components classified as Maintenance Rule category (a)(1) should not be eligible for test interval extension until they are no longer in (a)(1). Presently, the accumulator nitrogen supply vent valves are in (a)(1). Therefore, testing of these components will remain at the current Code frequency. In general, should a Maintenance Rule evaluation place a component with an extended IST in category (a)(1), the RI-IST program will test that component at the Code-prescribed frequency until such time that the component's performance history merits removal from (a)(1) status.

Summary

In summary, to blend deterministic and probabilistic information, the RI-IST Working Group deliberated on the limitations of PRA when it applied and made use of both plant-specific and generic information, as well as industry operating experience as applicable. At the end of the integrated decision-making process, every component eligible for test interval relaxation in the STP IST program was systematically reviewed and evaluated by the RI-IST Working Group and Expert Panel members.

The integrated decision-making process employed in support of this risk-informed application is assumed to be repeatable by another group consisting of members of similar technical knowledge. This position is based upon the availability of detailed technical bases for all sources of risk and the use of consistent ranking criteria applicable to both modeled and not modeled components.

3.0 CONCLUSIONS

The Executive Summary outlines the project scope, provides a succinct picture of STP's approach to addressing these issues, describes a basis for this approach, and identifies key project results and the most significant benefits derived from this project. The STP RI-IST team garnered insights from the experience of previous RI-IST projects and enhanced the proposed STP RI-IST program utilizing the latest regulatory insights and key experts within the STP organization as well as the industry at large. The result is a significantly enhanced program that more clearly delineates the importance of key plant equipment while optimizing the existing testing program to ensure acceptable equipment performance and safety margins are maintained. STP has confidence in these results based on insights from the PRA risk evaluations, equipment performance history, and comprehensive evaluations by key plant and industry experts.

The benefits of the STP integrated decision-making process -- inclusive of the RI-IST Working Group and plant Expert Panel -- may not be directly evident to the casual observer, but they are far reaching in their overall impact. The entire process not only improved the IST program, but as with any comprehensive cross-functional program, it raised the awareness across departmental boundaries, identified strengths and weaknesses in the IST and related programs, and reinforced the importance of teamwork within the organization. Key operations, maintenance, and engineering personnel involved in the RI-IST process have improved their understanding of the importance of equipment within the IST program.

4.0 NOTES AND REFERENCES

1. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant-specific Changes to the Licensing Basis," July 1998.

2. Regulatory Guide 1.175, "An Approach for Plant-specific, Risk-informed Decisionmaking: Inservice Testing," August 1998.

3. "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to the TU Electric Request to Implement a Risk-informed Inservice Testing Program at Comanche Peak Steam Electric Station (CPSES), Units 1 And 2, Docket Numbers 50-445 And 50-446."

4. "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to the Southern California Edison Request to Implement a Risk-informed Inservice Testing Program at San Onofre Nuclear Generating Station, Units 2 and 3, Docket Numbers 50-361 and 50-362."

5. Nuclear Regulatory Commission, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," Federal Register, Vol. 60, No. 158, August 16, 1995.

6. "Safety Evaluation by the Office of Nuclear Reactor Regulation [Related to the] Houston Lighting and Power Company South Texas Project, Units 1 and 2, Graded Quality Assurance Program, Docket Numbers 50-498 and 50-499."

7. "Safety Evaluation by the Office of Nuclear Reactor Regulation, Risk-informed Exemptions from Special Treatment Requirements, STP Nuclear Operating Company, South Texas Project Electric Generation Station, Units 1 and 2, Docket Nos. 50-498 and 50-499."

8. Containment isolation valves to be tested per 10 CFR 50, Appendix J, Option B account for less than 5% (27 components) of the Unit 1 IST components.

9. NRC Correspondence dated March 15, 1999, Inservice Testing Program Relief Request RR-17, South Texas Project, Units 1 and 2.

10. The RI-IST program study employs the results of the risk-informed GQA program study.

11. NRC's (Office of Nuclear Reactor Regulation) January 21, 1992 safety evaluation report on the Level I PSA submitted on April 14, 1989.

12. NRC's (Office of Nuclear Reactor Regulation) August 31, 1993 safety evaluation on the external events analysis in the Level 1 PSA submitted on April 14, 1989.

13. NRC's (Office of Nuclear Regulatory Research) June 27, 1995 staff evaluation of the Level 2 enhancements made to the 1989 PSA and submitted as the licensee's Individual Plant Examination (IPE) on August 28, 1992.

14. South Texas Project Electric Generating Station Level 2 Probabilistic Safety Assessment and

Individual Plant Examination, August 1992.

15. A Review of the South Texas Project Probabilistic Safety Analysis for Accident Frequency Estimates and Containment Binning, NUREG/CR-5606, August 1991.

16. Review of South Texas Project Units 1 and 2 Individual Plant Examination of External Events (IPEEE) Submittal NRC letter, dated 12/15/98.

17. The safe shutdown earthquake for STP is 0.1g.

18. Notice of Consideration of Issuance of Amendments - South Texas Project, Units 1 and 2 (Tac Nos. M92169 and M92170), Safety Evaluation Report of Diesel Generator Extended Allowed Outage Time, NRC letter dated February 2, 1996.

19. The Westinghouse Owners Group (WOG) Certification of the South Texas Project PRA is currently scheduled for April 2002

20. "Safety Evaluation by the Office of Nuclear Reactor Regulation [Related to the] Houston Lighting and Power Company South Texas Project, Units 1 and 2, Graded Quality Assurance Program, Docket Numbers 50-498 and 50-499," section 3.2.6.

21. Containment isolation valves to be tested per 10 CFR 50, Appendix J, Option B account for less than 5% (27 components) of the Unit 1 IST components.

22. All system level truncation levels are less than 1E-11 and only one systems analysis is equal to 1E-11.

23. Memorandum from Dr. William E. Vesely of SAIC to Mr. Mark Cunningham of NRC, "Reservations with ASME Risk-based Inservice Inspection and Testing," April 17' 1996.

24. NUREG/CR-6508, "Component Unavailability versus Inservice Test (IST) Interval: Evaluations of Component Aging Effects with Applications to Check Valves," developed by Oak Ridge National Laboratory for the NRC's Division of Engineering Technology Office of Nuclear Regulatory Research, July 1997.

25. E.V. Lofgren, et al., "Nuclear Power Plants Standby and Demand Stress Component Failure Modes: Methodology, Database, and Risk Implications," prepared by SAIC for US NRC Divisions of Systems Research Probabilistic Risk Analysis Branch, February 1992.

Attachment 2

RISK-INFORMED INSERVICE TESTING PROGRAM DESCRIPTION SUMMARY

RISK-INFORMED INSERVICE TESTING

PROGRAM DESCRIPTION SUMMARY

The document presents a proposed alternative to the ASME Section XI Inservice Testing Program at the South Texas Project. It is a risk-informed process which determines the safety significance and testing strategy of components in the ASME Section XI Inservice Testing (IST) Program, and identifies non-ASME IST components (pumps & valves) modeled in the Probabilistic Risk Assessment (PRA) determined to be High Safety Significant Components (HSSCs). The risk-informed inservice testing (RI-IST) process consists of the following elements:

1. Categorize components by Fussell-Vesely (FV) and Risk Achievement Worth (RAW) importance measures based on the STP Living PRA. (PRA Process)
2. Blend deterministic and probabilistic data to perform a final importance categorization of components as either RI-IST Low (Low), RI-IST Medium (Medium), or RI-IST High (High). (Integrated Decisionmaking Process - IDP)
3. Develop/Determine Test Frequencies and Test Methodologies for the ranked components. (Testing Philosophy)
4. Evaluate cumulative risk impact of new test frequencies and test methodologies to ensure risk reduction or risk neutrality. (Cumulative Risk Impact)
5. Develop an implementation plan. (Implementation)
6. Develop a performance monitoring plan for RI-IST Components. (Monitoring)
7. Develop a corrective action plan. (Corrective Action)
8. Perform periodic reassessments. (Periodic Reassessment)
9. Develop a methodology for making changes to the Risk-informed Inservice Testing (RI-IST) program. (Changes to RI-IST)

With these elements and their implementation, the key safety principle discussed in the Basis for Acceptance is maintained.

1.0 PRA PROCESS

PRA methodology facilitates determination of the risk significance of components based on end states of interest, such as core damage frequency (CDF) and release of radioactivity (e.g., large early release frequency (LERF)).

The PRA used to develop the importance measures is adequate for this application, and is complemented by the Integrated Decisionmaking Process (IDP), which includes an RI-IST Working Group and plant Expert Panel performance and review of the component categorization process, respectively. Evaluation of initiating events also includes loss of support systems and other special events such as Loss of Coolant Accident (LOCA), Steam Generator Tube Rupture (SGTR), Station Blackout (SBO), and Anticipated Transient Without Scram (ATWS).

The STP living PRA will be used to initially categorize components based on risk importance and also used to calculate changes in core damage frequency and large early release frequency. The initial categorization and change in CDF and LERF will be provided to the working group as part of the IDP. The quality of the Living PRA will be maintained under a formal PRA change and review process to ensure that the component importance measures and CDF/LERF calculations accurately reflect the as-built design and operation of STP.

The PRA will be periodically updated (See Section 8.0) to reflect the current plant design, procedures, and programs.

Component Ranking

Two figures of merit will be used to initially categorize components: Fussell-Vesely (FV) and Risk Achievement Worth (RAW). For the RI-IST Program, the following criteria will be used to initially rank components for review by the Integrated Decisionmaking Process (IDP).

Category	Criteria
RI-IST Rank High	RAW $\geq$ 100.0 OR FV $\geq$ 0.01 OR FV $\geq$ 0.005 and RAW $\geq$2.0
RI-IST Rank Medium (further evaluation required)	FV < 0.005 and 100.0> RAW $\geq$10.0
RI-IST Rank Medium	0.01 > FV $\geq$0.005 and RAW < 2.0 OR FV < 0.005 and 10.0 > RAW $\geq$2.0
RI-IST Rank Low	FV<0.005 and RAW<2

These CDF and LERF thresholds, coupled with the cumulative risk impact evaluation detailed in Section 4.0, ensure that the cumulative risk impact due to changes in test frequencies are within the acceptance guidelines of Regulatory Guides 1.174.

Methodology/Decision Criteria for PRA

The following describes a methodology that will be used to categorize components in the RI-IST when the program is reassessed. However, only those elements that are significantly affected by the model changes (e.g., design modifications or procedural changes) need to be reviewed in detail using this process. The scope of the review and the justification for it will be documented as part of the IDP. The following steps will be applied by the IDP:

1. Review FV and RAW importance measures for pumps and valves considered in the PRA against the classification criteria.
2. Review component importance measures to ensure that their bases are well understood and are consistent with the STPEGS specific levels of redundancy, diversity, and reliability.

PRA Limitations

To address limitations in the PRA, STP PRA analysts will apply the following treatments:

a) Address the sensitivity of the results to common cause failures (CCF), assuming all/none of the CCF importance is assigned to the associated component.

b) Evaluate other sensitivity studies (e.g., a study that evaluates the effects due to human action modeling). Identify/evaluate proceduralized operator recovery actions omitted by the PRA that can reduce the ranking of a component.

c) Consider industry history for particular IST components. Review such sources as NRC Generic Letters, Significant Operating Event Reports (SOERs), and Technical Bulletins and rank accordingly.

d) For components with high RAW and low FV, ensure that other compensatory measures are available to maintain the reliability of the component.

e) Identify and evaluate components whose performance shows a history of causing entry into limiting conditions for operation (LCO) conditions. To ensure that safety margins are maintained, consider retaining the ASME test frequency for these components.

Level II (LERF)

Consider components/systems that are potential contributors to large, early release. Determine LERF FV and RAW for components and/or determine which would have the equivalent of a high FV or low FV and high RAW with respect to LERF and rank accordingly. Also, in order to ensure that containment integrity continues to be maintained, consider:

- Containment isolation features that may not directly impact the value of LERF, and
- Interfacing systems LOCA that may provide a direct release path outside containment.

IST Components Not in the PRA

Review scenarios involving the "not-modeled" IST components to validate that the components are in fact low risk.

High-Risk PRA Components Not in the IST Program

- Identify, if any, other high risk pumps and valves (or, possibly non-Code components) in the PRA that are not in the IST program but should be tested commensurate with their importance.
- Determine whether current plant testing is commensurate with the importance of these components. If not, determine what test, e.g., the IST test, would be the most appropriate.

Other Considerations

Review the PRA to determine that sensitivity studies for cumulative effects and defense in depth have been adequately addressed in the determination of component importance factors.

2.0 Integrated Decisionmaking Process

The purpose of using the IDP is to confirm or adjust the initial risk ranking developed from the PRA results, and to provide a qualitative assessment based on engineering judgement and expert experience. This qualitative assessment compensates for limitations of the PRA, including cases where adequate quantitative data is not available.

The IDP uses deterministic insights, engineering judgement, experience, and regulatory requirements as detailed in this section. The IDP will review the initial PRA risk ranking, evaluate applicable deterministic information, and determine the final safety significance categories. The IDP considerations will be documented for each individual component to allow for future repeatability and scrutiny of the categorization process.

The scope of the IDP includes both categorization and application. The IDP is to provide deterministic insights that might influence categorization. The IDP will identify components whose performance justifies a higher categorization.

The IDP will determine appropriate changes to testing strategies. The IDP will identify compensatory measures for medium safety significant components, or justify the final categorization. The IDP will also concur on the test interval for components categorized as a Low Safety Significant Component (LSSC). The end product of the IDP will be components categorized as RI-IST Low, RI-IST Medium, or RI-IST High.

In making these determinations, the IDP ensures that key safety principles (namely defense-in-depth and safety margins), are maintained. It also ensures the changes in risk for both CDF and LERF are acceptable per the guidelines discussed in Section 1.0 above. The key safety principles are described below.

Defense in Depth

The STPEGS RI-IST program ensures consistent defense in depth by maintaining strict adherence to seven objectives of the defense in depth philosophy described in Regulatory Guides 1.174 and 1.175. The review and documentation of these objectives are an integral feature of the IDP for future changes to the program. Those objectives are:

1) A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. Multiple risk metrics, including CDF and LERF, will be used to ensure reasonable balance between risk end states (Objective 1).

2) No changes to the plant design or operations procedures will be made as part of the RI-IST program which either significantly reduces defense-in-depth, barrier independence or places strong reliance on any particular plant feature, human action, or programmatic activity (Objective 2, 5).

3) The methodology for component categorization --namely the selection of importance measures and how they are applied and understanding the basic reasons why components are categorized RI-IST Low, Medium, or High-- will be reviewed to ensure that redundancy and diversity are preserved as the more important principles. Component reliability can be used to categorize a component RI-IST Low or RI-IST Medium only when:

    a) plant performance has been good, and

    b) a compensatory measure or feedback mechanism is available to ensure adverse trends in equipment performance can be detected in a timely manner.

    A review will ensure that test frequency relaxation in the RI-IST program occurs only when the level of redundancy or diversity in the plant design or operation supports it. In this regard, all components that have significant contributions to common cause failure will be reviewed to avoid relaxation of requirements on those components with the lowest level of diversity within the system (Objective 3, 4).

4) Defenses against human errors are preserved by performing sensitivity studies. Sensitivity studies will be performed for human actions to ensure that components which mitigate the spectrum of accidents are not ranked low solely because of the reliability of a human action (Objective 6).

5) The intent of the General Design Criteria in 10CFRPart 50, Appendix A will be maintained (Objective 7).

<u>Other Considerations Related To Defense-In-Depth</u>

When the PRA does not explicitly model a component, function, or mode of operation, a qualitative method may be used to classify the component HSSC, MSSC, or LSSC and to determine whether a compensatory measure is required. The qualitative method is consistent with the principles of defense in depth because it preserves the distinction between those components which have high relative redundancy and those which have only high relative reliability.

Maintain Sufficient Safety Margin

The IDP will perform reviews consistent with Regulatory Guides 1.174 and 1.175 to ensure that sufficient safety margin is maintained when compared to the deterministic IST program. In performing this review, the IDP will consider such things as proposed changes to test intervals and, where appropriate, test methods. The IDP will ensure that the proposed compensatory measures, when required by the program, are effective in maintaining adequate safety margin. To enhance the safety margin, the IDP will also review PRA important components not in the current IST program for potential inclusion in the RI-IST program.

Categorization Guidelines

Working Group Structure and Role

The role of the RI-IST Working Group is crucial in ensuring that the results presented in this submittal are comprehensive. The Working Group not only considers the basis for the PRA risk measure for modeled components, but also qualitatively assesses the following for each component group:

- The degree to which component failure leads to an increase in the frequency of initiating events,
- The degree to which component failure leads to the failure of another safety system,
- The degree to which component failure causes a transient,
- The role of the component in the plant Emergency Operating Procedures (EOPs), and
- The role of the component in plant shutdown.

As part of the process, the Working Group authors a narrative basis to support the final RI-IST categorization of each component group.

The Working Group consists of members with expertise in the following disciplines:

- Power plant operations*,
- Plant maintenance*,
- PRA and nuclear safety analysis*,
- Systems engineering,
- Design basis engineering*,
- Safety analysis (Chapter 15)*,
- Quality assurance,
- Licensing, and
- Inservice testing (including ASME B&PV Code Section XI and ASME Code Cases)*.

*denotes voting members. Five voting members are required for quorum.

Periodic participation by a plant licensing expert and other component or system experts is on an as-required basis. Each core member of the Working Group shall have at least ten years experience in nuclear power and at least five years site-specific experience.

The RI-IST Working Group used plant knowledge, operating experience, and engineering judgment to perform the following tasks:

- Verify component functional failure modes
- Establish risk-informed categorizations for components not modeled in the PRA
- Assess or provide qualitative deterministic criteria
- Consider and/or provide insight concerning the component performance history. Specific attention was afforded to areas of poor or declining performance.
- Address all significant safety and operational concerns
- Validate component categorizations
- Resolve questions relative to PRA model completeness
- Resolve all questions raised during the review process

The RI-IST Working Group considers the following factors in addition to the combination of risk significance and deterministic insights discussed above:

- Important design basis functions not reflected in the risk categorizations
- Impact of PRA scope limitations, assumptions, and model simplifications, such as exclusion of shutdown states
- Importance of release states less severe than large early releases that are not explicitly reflected in the risk categorization scheme

The RI-IST Working Group also considers as part of their evaluation the uncertainties caused by:

- PRA model assumptions
- Common cause or common mode failure rates
- Treatment of support systems
- Level of definition of cutsets and cutset truncation
- Model assumptions relative to repair and restoration of failed equipment
- Human error rates used in the PRA
- Limitations in the meaning of importance measures

Based on the process outlined above, the Working Group makes a qualitative assessment of the RI-IST importance categories that were developed for the components using the PRA results and deterministic insights, plant-specific history, engineering judgements, and probabilistic risk analysis insights. The Working Group reviews the PRA component risk rankings, compares the PRA and IST functions to ensure consistency with plant design, and analyzes applicable deterministic information in its effort to resolve the final safety significance categorizations for all the IST components scrutinized.

Expert Panel Structure and Role

Subsequent to Working Group initial RI-IST categorization of components, the STP Expert Panel considers and ultimately validates the results of all Working Group activities and studies performed by the

IST project members. The Expert Panel consists of members with expertise in the areas of power plant operations, plant maintenance, PRA and nuclear safety analysis, design engineering, and quality assurance. The Expert Panel serves as the central point of decision-making for major technical issues and offers guidance to risk-informed IST project members in performing their work. Because STP requires that the Expert Panel perform this very function for all plant risk-informed programs, consistency in decision bases and management of commitments across plant programs is assured.

Modeled Components/Functions

RI-IST Rank High	RAW ≥ 100.0 OR FV ≥ 0.01 OR FV ≥ 0.005 and RAW ≥2.0
RI-IST Rank Medium (further evaluation required)	FV < 0.005 and 100.0> RAW ≥10.0
RI-IST Rank Medium	0.01 > FV ≥0.005 and RAW < 2.0 OR FV < 0.005 and 10.0 > RAW ≥2.0
RI-IST Rank Low	FV<0.005 and RAW<2

For modeled components/functions with a FV > 0.01, or a FV > .005 and a RAW > 2, or a RAW greater than 100, the IDP confirms the component categorization as RI-IST High.

For modeled components/functions with a FV between 0.01 and 0.005 and a RAW < 2, or a FV < 0.005 and a RAW between 2 and 100, the IDP will rank the component as RI-IST Medium. The component may effectively be considered RI-IST Low, provided a compensatory measure exists that ensures operational readiness and the component's performance is acceptable. If a compensatory measure is not available or the component has a history of poor performance, the component will not be considered for test interval extension and will be considered for potential test method enhancement.

For modeled components/functions with a FV < 0.005 and a RAW < 2.0, the component will be categorized as RI-IST Low, provided the component's performance has been acceptable. Components with a history of poor performance will only be considered for test interval extension if a compensatory measure is identified to ensure operational readiness.

Non-Modeled Components/Functions

For components not modeled or the safety function not modeled in the PRA, the categorization is as follows:

- If the sister train is modeled, then the component assumes that final categorization.
- If the component is implicitly modeled in the PRA, the FV and RAW are estimated and the deliberation is as discussed for modeled components/functions.

- If the component is not implicitly modeled, the component performance history will be reviewed. For acceptable performance history the component will be categorized as RI-IST Low. For poor performance history, a compensatory measure will be identified to ensure operational readiness and the component will be categorized as RI-IST Low. If no compensatory measures are available, the component will be not be considered for test interval extension until performance is improved.

Documentation

Documentation of the IDP will be available for review at the plant site. The basis for risk ranking and component grouping will be entered in the IST data system.

3.0 Testing Philosophy

Motor-Operated Valves (MOVs)

RI-IST High

Diagnostic testing will be performed in accordance with NRC Generic Letter 89-10 and 96-05 commitments as described in the Joint Owners Group Periodic Verification Program (JOG PV Program). Stroke time testing will be replaced by exercising all valves in each group at least once per refueling cycle and diagnostically testing these MOVs in accordance with STP commitments to the JOG PV Program. MOVs with safety functions not tested in accordance with the above GNL requirements will be tested per 10CFR50.55a at quarterly, cold shutdown, or refueling interval based on the practicability of testing.

RI-IST Medium

Diagnostic testing will be performed in accordance with NRC Generic Letter 89-10 and 96-05 commitments as described in the Joint Owners Group Periodic Verification Program (JOG PV Program). Stroke time testing will be replaced by exercising all valves in each group at least once per refueling cycle and diagnostically testing these MOVs in accordance with STP commitments to the JOG PV Program. MOVs with safety functions not tested in accordance with the above GNL requirements will be tested per 10CFR50.55a, except, based on evaluation of design, service condition, and performance history, and compensatory actions, at a test frequency not to exceed 6 years (plus a 25% margin based on a 2-year interval) and exercised at least once during a refueling cycle.

RI-IST Low

Diagnostic testing will be performed in accordance with NRC Generic Letter 89-10 and 96-05 commitments as described in the Joint Owners Group Periodic Verification Program (JOG PV Program). Stroke time testing will be replaced by exercising all valves in each group at least once per refueling cycle and diagnostically testing these MOVs in accordance with STP commitments to the JOG PV Program. MOVs with safety functions not tested in accordance with the above GNL requirements will be tested per 10CFR50.55a, except, based on evaluation of design, service condition, and performance history, at a test frequency not to exceed 6 years (plus a 25% margin based on a 2 year frequency) and exercised at least once during a refueling cycle.

Seat leakage testing, if required, will be per 10CFR50.55a.

STP will ensure procedurally that the potential benefits (such as identification of decreased force output and increased force requirements) and potential adverse effects (such as accelerated degradation due to aging or valve damage) are considered when determining the appropriate testing for each MOV.

RI-IST program and MOV trend procedures will contain guidance to ensure performance and test experience from previous tests are evaluated to justify the periodic verification interval.

STP will develop and proceduralize a method to determine an MOV test interval that is based on IDP final risk ranking, available valve margin, and valve performance history. The method will be comprised of an evaluation of risk ranking, relative margin, and group as well as individual valve performance.

The result of the evaluation determines the testing interval with the most frequent testing interval applied to high risk, low margin valves with poor, or questionable performance history. Stepwise increases in interval out to the maximum allowable interval depend on the combination of risk rank, margin, and performance history.

Relief Valves

Testing of relief valves will continue to be conducted in accordance with 10CFR50.55a (OM-1) with no change in test interval. STP believes that relief valve performance, as a whole, does not warrant interval extension. In the future, should performance history change, STP will rank valves per the IDP and extend intervals accordingly. The initial testing strategy will be:

RI-IST High

Testing will be performed in accordance with 10CFR50.55a.

RI-IST Medium

Testing will be performed in accordance with 10CFR50.55a.

RI-IST Low

Testing will be performed in accordance with 10CFR50.55a.

Check Valves

RI-IST High

Testing will be performed in accordance with 10CFR50.55a.

RI-IST Medium

Testing will be performed in accordance with 10CFR50.55a except, based on evaluation of design, service condition, performance history, and compensatory actions, the test interval may be extended not to exceed 6 years (plus a 25% margin based on a 2-year frequency).

RI-IST Low

Testing will be performed in accordance with 10CFR50.55a except, based on evaluation of design, service condition, and performance history, the test interval may be extended not to exceed 6 years plus a 25% margin based on a 2-year frequency.

RI-IST High, RI-IST Medium, and RI-IST Low check valves at STP are included in the Check Valve Program (CVP), which has been developed to provide confidence that check valves will perform as designed. Station procedure(s) establish test/exam frequencies, methods, and acceptance criteria and provide performance-monitoring requirements for check valves in the CVP. Check valves in the CVP include check valves that are in the IST program, check valves identified as susceptible to unusually high wear, fatigue, or corrosion, and special valves used for personnel safety such as those in the breathing air system. The CVP includes approaches for identification of existing and incipient check valve failures using non-intrusive (e.g., radiography, acoustic emission (AE), magnetic flux (MF), and/or ultrasonic examination (UT) testing methods) and disassembly examination. Test data will be used (e.g., trended as appropriate) to provide confidence that check valves in the CVP will be capable of performing their intended function until the next scheduled test activity. Check valves may be added to or deleted from the CVP based on non-intrusive testing, disassembly examination results, component replacement, or site maintenance history.

The CVP is assessed and updated as appropriate with new design and operational information, and incorporates any applicable site or industry lessons learned.

Air Operated Valves (AOVs)

RI-IST High

Testing will be performed in accordance with 10CFR50.55a.

RI-IST Medium

Testing will be performed in accordance with 10CFR50.55a, except based on evaluation of design, service condition, performance history, and compensatory actions, the test interval may be extended not to exceed 6 years (plus a 25% margin based on a 2-year interval). Additionally, RI-IST Medium AOVs will be stroked at least once during each operating cycle.

RI-IST Low

Testing will be performed in accordance with 10CFR50.55a, except based on evaluation of design, service condition, and performance history, the test interval may be extended not to exceed 6 years (plus a 25% margin based on a 2-year interval). Additionally, RI-IST Low AOVs will be stroked once during the operating cycle.

STP Nuclear Operating Company has committed to work with the Joint Owners Group for Air Operated Valves (JOG AOV) to develop an enhanced AOV testing program. The intent of this program is to specify AOV Program requirements to provide assurance that AOVs are capable of performing their intended safety-significant or risk-significant functions. Elements of the proposed program include establishing scoping and categorization, setpoint control, design basis review, testing, preventative maintenance, training, feedback, tracking and trending AOV performance. STP's current testing program meets or exceeds the current JOG AOV testing requirements for components within the IST program. Design basis evaluations will be performed for AOV Program Category 1 valves. These evaluations will check the available capability margin versus the required design-bases conditions to ensure adequate margin does indeed exist. The JOG AOV Program does not include dampers (except in hard pipe), hydraulic, or solenoid valves (unless in the AOV circuits).

The current STP AOV program is assessed and updated as appropriate with new design and operational information, and incorporates any applicable site or industry lessons learned.

Hydraulic Valves (HOVs), Solenoid Valves (SOVs), and Others (Manual Valves, etc.)

STP proposes to test these valves in accordance with 10CFR50.55a (OM Part 10) with the exception that the test frequency will be in accordance with the component risk categorization defined below:

RI-IST High

Testing will be performed in accordance with 10CFR50.55a.

RI-IST Medium

Testing will be performed in accordance with 10CFR50.55a except, based on evaluation of design, service condition, performance history, and compensatory actions, the test interval may be extended not to exceed 6 years (plus a 25% margin based on a 2-year frequency). Additionally, RI-IST Medium HOVs and SOVs will be stroked once during the operating cycle.

RI-IST Low

Testing will be performed in accordance with 10CFR50.55a except, based on evaluation of design, service condition, and performance history, the test interval may be extended not to exceed 6 years (plus a 25% margin based on a 2-year interval). Additionally, RI-IST Low HOVs and SOVs will be stroked once during the operating cycle.

Pumps

Pumps will be tested in accordance with 10CFR50.55a (OM Part 6) with the exception that the test frequency may be in accordance with the component risk categorization defined below:

RI-IST High

Testing will be performed in accordance with 10CFR50.55a.

RI-IST Medium

Testing will be performed in accordance with 10CFR50.55a except, based on evaluation of design, service condition, performance history, and compensatory actions, the test interval may be extended not exceed 6 years (plus a 25% margin based on a 2-year interval).

RI-IST Low

Testing will be performed in accordance with 10CFR50.55a except, based on evaluation of design, service condition, and performance history, the test interval may be extended not to exceed 6 years (plus a 25% margin based on a 2-year interval).

All pumps will receive periodic thermography of their driver, lube oil analysis, alignment checks performed following major pump maintenance (using vibration analysis methods to confirm alignment), motor current testing (when the motor current testing program is implemented), vibration monitoring (required by the current Code). Additional tests (e.g., thermography of the driver, or motor current testing[26]) are predictive in nature and involve trending of parameters. This augmented testing program for pumps provides reasonable assurance that adequate pump capacity margin exists such that pump operating characteristics over time do not degrade to a point of insufficient margin before the next scheduled test activity.

4.0 CUMULATIVE RISK IMPACT

As part of the IDP review, the change in CDF and LERF will be calculated. The change in CDF and LERF will account for (but may not be limited to) changes in component availability, reliability, test intervals, and implemented test strategies (e.g., staggered testing, enhanced testing). The change in CDF and LERF will also be calculated for proposed changes to component test strategies and test intervals and their impact on component reliability, initiating event frequency and common-cause failure probabilities. This review ensures that the incremental CDF and LERF change of 1) the implemented risk-informed program from the deterministic IST program and 2) the risk-informed program until the next IDP review (two fuel cycles) remain within the risk change guidelines of Regulatory Guides 1.174 and 1.175.

5.0 IMPLEMENTATION

Implementation of the RI-IST -- including components ranked either RI-IST Low or RI-IST Medium -- will consist of grouping components and then staggering the testing of the group over the test frequency.

Grouping:

Components will generally be grouped based on:

- System
- Component type (MOV, AOV, Check Valve, etc.)
- Manufacturer
- Size
- Style (globe, gate, swing check, tilt disk, etc.)
- Application (pump discharge, flow path, orientation, etc).

The population of the group will be dependent on:

- Total population available

[26] Both driver thermography and motor current testing are currently in the early stages of implementation at STP.

- Maintaining current testing schedule

Grouping components in this manner and testing on a staggered basis over the test interval reduces the importance of common cause failure modes since at least one valve in the group is tested on a subinterval determined by the number of valves in the group.

Testing of components within the defined group will be staggered over the test interval, typically 6 years. Testing will be scheduled on regular sub-intervals over the test interval to ensure all components in the group are tested at least once during the test interval, the same component is not tested repeatedly, while deferring others in the group, and not all components are tested at one time. The staggering allows the trending of components in the group to ensure the test frequency selected is appropriate. A test interval extension of 25% of the fundamental stagger interval (i.e. 1 refueling cycle or 2 years) accommodates operational circumstances that may interfere with establishing the plant conditions to meet the baseline test schedule. For component groups that are insufficient in size to test one component each refueling cycle, the implementation of interval extensions will be accomplished in a step-wise manner.

Additionally, both STP units are essentially identical and the IST integrated decision-making process considered operational experience and maintenance history from both units. Following the guidance of NUREG-1482 for grouping of components, valves with like design and construction in both units can be grouped for staggered testing as described above.

6.0 PERFORMANCE MONITORING OF RI-IST COMPONENTS

In addition to the specific inservice testing proposed for each component group discussed in Section 3.0 above, the following additional monitoring for each component group is currently in place per existing site procedures. The additional performance monitoring activities listed by component type are applicable to all components regardless of individual ranking (RI-IST High, RI-IST Medium, or RI-IST Low).

The proposed monitoring plan is sufficient to detect component degradation in a timely manner. Further, the monitoring activities identified for each component group ensure that the following criteria are met:

- Sufficient tests are conducted to provide meaningful data.
- The inservice tests are conducted such that the probability of detecting incipient degradation is high.
- Appropriate parameters are trended to provide reasonable assurance that the component will remain operable over the test interval.

The proposed performance-monitoring plan is sufficient to ensure that degradation is not significant for components placed on an extended test interval, and that failure rates assumed for these components will not be significantly compromised. The proposed performance monitoring, when coupled with STP's corrective action program (discussed in Section 7), ensures corrective actions are taken and timely adjustments are made to individual component test strategies where appropriate.

Components that do not warrant test frequency extension based on limited, poor, or marginal performance histories will be monitored through the Corrective Action and Integrated Decisionmaking Processes and reviewed during the program periodic reassessment as described in Section 8.

The STP RI-IST Program will be reassessed at a frequency not to exceed once every other refueling outage (approximately 3 years), following Unit 1 refueling outage, to reflect changes in plant configuration, component performance test results, industry experience, and other inputs to the process. Configuration changes will be assessed in concert with the current design change process. Therefore, the monitoring process for RI-IST is adequately coordinated with existing programs (e.g., Corrective Action Program, Maintenance Rule monitoring, and design change process) for monitoring component performance and other operating experience on this site and, where appropriate, throughout the industry. Although the monitoring of reliability and unavailability goals for some operating and standby systems/trains is required by the Maintenance Rule, it alone will not be relied upon to ensure operational readiness of components in the RI-IST program. The STP Corrective Action Program requires timely operability assessment for component performance issues detected outside the auspices of the IST program. This process, coupled with the evaluations performed under the Maintenance Rule in concert with IST trending, ensures continued operational readiness of RI-IST components. The individual condition monitoring points for each component type are governed by site procedures and the 10CFR50.59 change process.

Preventative maintenance activities are dictated by the individual component procedures. Intervals range from one to five refueling cycles depending on component type, application, and individual performance history. The periodicity may be altered as accumulated data and industry experience warrant via site procedures, the IDP, and the 10CFR50.59 change process. The specific inspection points may vary as dictated by inspection and diagnostic test results. The preventive maintenance activities currently include the items listed below:

Motor-Operated Valves (MOVs)

- Actuator electrical visual inspections
    - Limit switch assemblies
    - Torque switch assemblies
    - Wiring
    - Motor T-drains
    - Motor condition
- Actuator mechanical visual inspection
    - Inspect fasteners, gaskets, and packing
    - Inspect stem protective cover
    - Inspect for lubrication leaks
    - Document other observable damages
- Actuator lubrication inspection
    - Inspect for lubrication condition
    - Add lubrication to stem
    - Lubricate main gearbox
    - Lubricate motor gearbox

- Inspect stem nut for tightness and staking
- Other activities
    - Perform hand wheel operation
    - Visual inspection for gross irregularities, upper bearing housing cover for warping on SMB-000,
    - Verify/tighten actuator mounting bolts, anti-lock rotation plate jam nuts
    - Monitor stem nut thread condition

Relief Valves

- Test results trended
- New valves tested prior to installation
- Valves set as close to nominal as practical

Check Valves

- Combination of acoustic, magnetic, and/or ultrasonic testing methods are used as appropriate
- Data retrieved from these methods will be compared with previous results and the differences evaluated
- Open and close exercise testing
- Check valve disassembly inspections are performed where other testing is not practicable
- Leak rate testing is performed by 10CFR50, Appendix J program where appropriate
- Leak testing for check valve closed exercise testing where appropriate

Air-Operated Valves (AOVs)

AOV preventative maintenance activities are currently scheduled not to exceed 5 fuel cycles for Category 2 valves and 4 fuel cycles for Category 1 valves. This initial periodicity may be altered as accumulated data and industry experience warrant as described below. The specific inspection points may vary as dictated by inspection and diagnostic test results. Initial intervals as well as the specific points monitored may be adjusted per station procedures and the 10CFR50.59 process. The preventive maintenance activities initially include the items listed below:

- Routine overhauls (scheduled as noted for Category 1 & 2 above) that include:
    - Disassembly, cleaning, inspection
    - Replacement of elastomers
    - Replacement of air filter / pressure regulator assembly
    - Re-assembly and testing
    - Response time testing
    - Diagnostic testing as outlined below.
- Valves exposed to extreme environmental conditions will have repetitive maintenance orders for actuator replacement consistent with the service conditions.
- Positioner PMs consist of the following:

- Removal disassembly, cleaning, inspection
- Parts replacement as required
- Reassembly and test

- Static diagnostic testing performed following valve or actuator overhaul (Preventive Maintenance) or corrective maintenance that could impact valve function, or as requested.
- Diagnostic testing of the following testing parameters as applicable
  - Bench set
  - Maximum available pneumatic pressure
  - Seat load
  - Spring rate
  - Stroke time
  - Actual travel
  - Total friction
  - Minimum pneumatic pressure required to accomplish the safety function(s) of the valve assembly (under development)
  - Pneumatic pressure at appropriate point in operation
  - Set point of pressure switch(s), relief valve, regulator, etc
- Others as dictated by the specific valve/actuator style and application.

Pumps

- Margin to safety limit deviations – head curves
- Lube oil analysis
- Alignment checks
- Motor current testing
- Vibration monitoring
- Thermography

7.0 CORRECTIVE ACTION

When an RI-IST Low or RI-IST Medium component on the extended test interval fails to meet established test criteria, corrective actions will be taken in accordance with the STP corrective action program as described below for the RI-IST.

For all components not meeting the acceptance criteria, a Condition Report (CR) will be generated. This document initiates the corrective action process. A CR may result from activities other than IST that identifies degradation in performance.

The initiating event could be any other indications that the component is in a non-conforming condition. The unsatisfactory condition will be evaluated to:

a) Determine the impact on system operability since the previous test.
b) Review the previous test data for the component and all components in the group.

c) Perform an apparent cause analysis and/or a root cause analysis as applicable.

d) Determine if this is a generic failure. If it is a generic failure whose implications affect a group of components, initiate corrective action for all components in the affected group.

e) Initiate corrective action for failed IST components.

f) Evaluate the adequacy of the test interval. If a change is required, review the IST test schedule and change as appropriate.

The results of component testing will be provided to and reviewed by the PRA group for potential impact to a PRA model update. The PRA model will be updated as necessary with changes tracked and documented per the PRA Change Process Program.

For an emergent plant modification, any new IST component added will initially be included at the current Code of Record test frequency. Only after evaluation of the component through the RI-IST Program (i.e., PRA model update if applicable and IDP review) will this be considered RI-IST Low or RI-IST Medium with an extended test interval.

8.0 PERIODIC REASSESSMENT

As a living process, components will be reassessed at a frequency not to exceed every other refueling outage (approximately 3 years based on Unit 1 refueling outages) to reflect changes in plant configuration, component performance test results, industry experience, and other inputs to the process. The RI-IST reassessment will be completed within 9 months of completion of the outage.

Part of this periodic reassessment will be a feedback loop of information to the PRA. This will include information such as components tested since the last reassessment, number and type of tests, number of failures, corrective actions taken including generic implication, and changed test frequencies. Once the PRA has been reassessed, the information will be brought back through the IDP for deliberation and confirmation of the existing lists of RI-IST High components, RI-IST Medium components, and RI-IST Low components, or modification of these lists based on the new data, if required. As part of the IDP, confirmatory measures previously used to categorize components as RI-IST Low, as well as compensatory measures used to justify the extension of RI-IST Medium components, will be validated.

During the periodic reassessment RI-IST Low and RI-IST Medium components whose performance history did not justify extension will be reviewed. The review will focus on the adequacy and effectiveness of corrective actions, as well as the performance of similar components in similar applications. If the Working Group judges the performance warrants a test interval extension based on the combination of risk metrics, available margin, and successive satisfactory performance, then with Working Group consensus the test interval may be adjusted.

Additionally, the maximum test interval for each component or component group will be verified or modified as dictated by the IDP.

9.0 CHANGES TO RI-IST

Changes to the process described above (such as acceptance guidelines used for the IDP) as well as changes in test methodology issues that involve deviation from NRC endorsed Code requirements, NRC endorsed Code Case, or published NRC guidance are subject to NRC review and approval prior to implementation. Other changes using the process detailed above (such as relative ranking, risk categorization, and grouping) are subject to site procedures and the associated change process pursuant to 10CFR50.59. STP will periodically submit changes to the NRC for their information.

Attachment 3

PUMP AND VALVE LISTS FOR 2ND 10-YEAR INTERVAL

This attachment includes separate reports that provide the information normally submitted as the IST Plan document for the update requirement. The first report is titled IST Valve Groups and it lists all the IST scoped valves which are grouped by like components as described in the Risk Informed Inservice Testing Program Description. As a result of the 10-year update to the OMa-1988 Code, this list also includes the relief valves which are now scoped in the IST program based on the requirements of the OM-1987 edition of Part 1.

The second report of this attachment is the listing of the testing requirements by group. This report shows the IST rank as determined by the Integrated Decisionmaking Process, the frequency tested under the previous edition of the Code, and the resulting risk informed test frequency. The table below provides a description of the frequency codes that are used in this report. Where applicable, a reference (i.e. CSJ-01) is added to indicate that the frequency is based on a cold shutdown or refueling outage justification.

IST FREQUENCY CODES			
Q	Once per Quarter	30MO	Every 30 months
CS	At Cold Shutdown	3YR	Every 3 years
2Y	Every 2 Years	54MO	Every 54 months
RF	At Refueling	5YR	Every 5 years
R	Every 18 months	6YR	Every 6 years
6M	Every 6 months	36MO	Every 36 months
App J	Tested per Appendix J Option B		

The next report is a list of the ASME pumps included in the IST scope. This report shows the pumps divided in the groups for staggered testing. The pump safety function and the IST rank are displayed. Again, the previous frequency and the resulting risk-informed test frequency are shown.

Finally, the last report provides the cases where STP is taking exception to the code requirements for RI-IST High rank components. These activities cannot be performed during normal power operations. The reasons for the testing exceptions and the proposed testing requirements are described. The report also includes the relief requests proposed by STP for situations where the ASME Code cannot be satisfied.

IST Valve Groups

GROUP	GROUP DESCRIPTION						VALVE DATA			VALVE POSITIONS		
	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
AF01	Auxiliary Feedwater Supply to Steam Generator Inside Cntmt Isolation Check Valves											
	2S141TAF0120	A	5S141F00024	D-1	2	C	8	CHECK	SELF	CLOS	N/A	O
	2S141TAF0121	A	5S141F00024	C-1	2	C	8	CHECK	SELF	CLOS	N/A	O
	2S141TAF0122	A	5S141F00024	H-1	2	C	8	CHECK	SELF	CLOS	N/A	O
	2S141TAF0119	A	5S141F00024	F-1	2	C	8	CHECK	SELF	CLOS	N/A	O
AF02	Auxiliary Feedwater Supply to Steam Generator Outside Cntmt Isolation Stop Check MOVs											
	2S141TAF0085	A	5S141F00024	B-2	2	B/C	4	STOP C	MOTOR	CLOS	FAI	O/C
	2S141TAF0065	A	5S141F00024	D-2	2	B/C	4	STOP C	MOTOR	CLOS	FAI	O/C
	2S141TAF0048	A	5S141F00024	F-2	2	B/C	4	STOP C	MOTOR	CLOS	FAI	O/C
	2S141TAF0019	A	5S141F00024	G-2	2	B/C	4	STOP C	MOTOR	CLOS	FAI	O/C
AF03	Auxiliary Feedwater Supply to Steam Generator Flow Regulating MOVs											
	3S141ZAF7524	A	5S141F00024	D-4	3	B	4	GLOBE	MOTOR	OPEN	FAI	O
	3S141ZAF7525	A	5S141F00024	F-4	3	B	4	GLOBE	MOTOR	OPEN	FAI	O
	3S141ZAF7523	A	5S141F00024	B-4	3	B	4	GLOBE	MOTOR	OPEN	FAI	O
	3S141ZAF7526	A	5S141F00024	H-3	3	B	4	GLOBE	MOTOR	OPEN	FAI	O
AF04	Auxiliary Feedwater Turbine Trip and Trottle Valve (MS0514)											
	3S141XMS0514	A	5R169F00024	F-6	3	B	4	GLOBE	MOTOR	CLOS	FAI	O/C
AF05	Main Steam to Auxiliary Feedwater Turbine Warm-up Valve											
	D1AFFV0143	A	5R169F00024	G-8	2	B	1	GLOBE	SOLENO	CLOS	CLOS	O/C
AF06	Auxiliary Feedwater Pump Discharge Cross-Tie Valves											
	A1AFFV7517	A	5S141F00024	F-5	3	B	4	GLOBE	AIR	OPEN	CLOS	O/C
	B1AFFV7516	A	5S141F00024	D-5	3	B	4	GLOBE	AIR	OPEN	CLOS	O/C
	C1AFFV7515	A	5S141F00024	B-5	3	B	4	GLOBE	AIR	OPEN	CLOS	O/C
AF07	Auxiliary Feedwater Auto Recirc Valves											

GROUP	GROUP DESCRIPTION / TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
							VALVE DATA			VALVE POSITIONS		
	3S141TAF0011	A	5S141F00024	H-5	3	C	4	CHECK	SELF	CLOS	N/A	O
	3S141TAF0036	A	5S141F00024	F-6	3	C	4	CHECK	SELF	CLOS	N/A	O
	3S141TAF0058	A	5S141F00024	D-6	3	C	4	CHECK	SELF	CLOS	N/A	O
	3S141TAF0091	A	5S141F00024	B-6	3	C	4	CHECK	SELF	CLOS	N/A	O
AF08	Main Steam to AF Turbine Suction Stop Check MOV (MS0143)											
	2S141TMS0143	A	5S141F00024	H-8	2	B	4	STOP C	MOTOR	OPEN	FAI	O/C
AF09	Auxiliary Feedwater Pump Discharge Cross-Tie Valve (D train)											
	D1AFFV7518	A	5R169F00024	G-4	3	B	4	GLOBE	AIR	OPEN	CLOS	O/C
AP01	RCS Hot Leg Sample to PASS Lab OCIVs											
	B1APFV2455A	A	5Z549Z47501	E-7	2	A	1	GATE	SOLENO	CLOS	CLOS	C
	B1APFV2455	A	5Z549Z47501	E-7	2	A	1	GATE	SOLENO	CLOS	CLOS	C
AP02	Cntmt Normal Sump to PASS Lab OCIVs											
	A1APFV2453	A	5Z549Z47501	G-7	2	A	1	GATE	SOLENO	CLOS	CLOS	C
AP03	RHR Sample to PASS Lab OCIVs											
	A1APFV2454	A	5Z549Z47501	F-7	2	A	1	GATE	SOLENO	CLOS	CLOS	C
AP04	PASS Waste Collection Unit Return to Pressurizer Relief Tank OCIV											
	C1APFV2458	A	5Z549Z47501	C-3	2	A	1	GATE	SOLENO	CLOS	CLOS	C
AP05	Containment Air Sample Supply and Return to PASS Lab OCIVs											
	C1APFV2457	A	5Z549Z47501	H-2	2	A	1	GATE	SOLENO	CLOS	CLOS	C
	C1APFV2456	A	5Z549Z47501	D-7	2	A	1	GATE	SOLENO	CLOS	CLOS	C
BA01	Breathing Air System Inside Cntmt Isolation Check Valve											
	2Q121TBA0006	P	5Q129F05044	H-4	2	A/C	1	CHECK	SELF	CLOS	N/A	C
BA02	Breathing Air System Outside Cntmt Isolation Manual Valve											
	2Q121TBA0004	P	5Q129F05044	G-4	2	A	1	BALL	MANUAL	CLOS	N/A	C
CC01	Thermal Relief for Penetration M-40 CCW return for the RCPs											
	2R201TCC0446	A	5R209F05021	B-1	2	A/C	1	CHECK	SELF	CLOS	N/A	O/C
CC02	CCW Supply to the RCP Thermal Barriers (Double inlet check valves)											

GROUP	GROUP DESCRIPTION / TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	3R201TCC0327	A	5R209F05021	B-8	3	C	2	CHECK	SELF	OPEN	N/A	O/C
	3R201TCC0321	A	5R209F05021	E-5	3	C	2	CHECK	SELF	OPEN	N/A	O/C
	3R201TCC0346	A	5R209F05021	E-8	3	C	2	CHECK	SELF	OPEN	N/A	O/C
	3R201TCC0363	A	5R209F05021	B-5	3	C	2	CHECK	SELF	OPEN	N/A	O/C
	3R201TCC0756	A	5R209F05021	E-4	3	C	2	CHECK	SELF	OPEN	N/A	O/C
	3R201TCC0758	A	5R209F05021	E-7	3	C	2	CHECK	SELF	OPEN	N/A	O/C
	3R201TCC0759	A	5R209F05021	B-8	3	C	2	CHECK	SELF	OPEN	N/A	O/C
	3R201TCC0757	A	5R209F05021	B-5	3	C	2	CHECK	SELF	OPEN	N/A	O/C
CC03	Penetration M-40 CCW return for the RCPs											
	D1CCFV4493	A	5R209F05021	H-1	2	A	12	BUTTER	AIR	OPEN	CLOS	C
CC04	RHR Heat Exchanger - CCW Outlet Valves											
	B1CCFV4548	A	5R209F05018	G-2	3	B	16	BUTTER	AIR	CLOS	CLOS	O
	A1CCFV4531	A	5R209F05017	G-2	3	B	16	BUTTER	AIR	CLOS	CLOS	O
	C1CCFV4565	A	5R209F05019	G-2	3	B	16	BUTTER	AIR	CLOS	OPEN	O
CC05	Common Suction Header Isolation Valves (Trains A, B, & C) MOVs											
	3R201TCC0052	A	5R209F05020	C-7	3	B	24	BUTTER	MOTOR	EITH	FAI	O/C
	3R201TCC0132	A	5R209F05020	C-7	3	B	24	BUTTER	MOTOR	EITH	FAI	O/C
	3R201TCC0192	A	5R209F05020	B-7	3	B	24	BUTTER	MOTOR	EITH	FAI	O/C
CC06	Common Supply Header Isolation Valves (Trains A, B, & C)											
	3R201TCC0316	A	5R209F05020	F-7	3	B	24	BUTTER	MOTOR	EITH	N/A	O/C
	3R201TCC0312	A	5R209F05020	E-7	3	B	24	BUTTER	MOTOR	EITH	N/A	O/C
	3R201TCC0314	A	5R209F05020	E-7	3	B	24	BUTTER	MOTOR	EITH	N/A	O/C
CC07	CCW Heat Exchanger Outlet MOVs (Trains A, B, and C)											
	3R201TCC0645	A	5R209F05018	B-5	3	B	24	BUTTER	MOTOR	OPEN	FAI	O
	3R201TCC0643	A	5R209F05017	B-5	3	B	24	BUTTER	MOTOR	OPEN	FAI	O

GROUP	GROUP DESCRIPTION TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	VALVE DATA Size	Type	Actuator	VALVE POSITIONS Normal	Failsafe	Safety Func.
	3R201TCC0647	A	5R209F05019	B-5	3	B	24	BUTTER	MOTOR	OPEN	FAI	O
CC08	CCW Heat Exchanger Bypass MOVs (Trains A, B, and C)											
	3R201TCC0646	A	5R209F05019	A-6	3	B	16	BUTTER	MOTOR	CLOS	FAI	O/C
	3R201TCC0644	A	5R209F05018	A-6	3	B	16	BUTTER	MOTOR	CLOS	FAI	O/C
	3R201TCC0642	A	5R209F05017	A-6	3	B	16	BUTTER	MOTOR	CLOS	FAI	O/C
CC09	CCW return from the RCFCs, Inside Containment Isolation Valves (Trains A, B, and C)											
	2R201TCC0147	A	5R209F05018	C-4	2	A	14	BUTTER	MOTOR	OPEN	FAI	O/C
	2R201TCC0068	A	5R209F05017	C-4	2	A	14	BUTTER	MOTOR	OPEN	FAI	O/C
	2R201TCC0208	A	5R209F05019	D-4	2	A	14	BUTTER	MOTOR	OPEN	FAI	O/C
CC09A	CCW return from the RCFCs, Outside Containment Isolation Valves (Trains A, B, and C)											
	2R201TCC0210	A	5R209F05019	D-4	2	A	14	BUTTER	MOTOR	CLOS	FAI	O/C
	2R201TCC0148	A	5R209F05018	D-4	2	A	14	BUTTER	MOTOR	CLOS	FAI	O/C
	2R201TCC0069	A	5R209F05017	D-4	2	A	14	BUTTER	MOTOR	CLOS	FAI	O/C
CC10	CCW Supply (OCIV) to RHR Pump and Heat Exchanger - Trains A, B, and C											
	2R201TCC0122	A	5R209F05018	E-2	2	A	16	BUTTER	MOTOR	OPEN	FAI	O/C
	2R201TCC0182	A	5R209F05019	F-1	2	A	16	BUTTER	MOTOR	OPEN	FAI	O/C
	2R201TCC0012	A	5R209F05017	E-2	2	A	16	BUTTER	MOTOR	OPEN	FAI	O/C
CC11	CCW Supply (OCIV) to Reactor Containment Fan Coolers - Trains A, B, and C											
	2R201TCC0197	A	5R209F05019	D-2	2	A	14	BUTTER	MOTOR	CLOS	FAI	O/C
	2R201TCC0136	A	5R209F05018	D-2	2	A	14	BUTTER	MOTOR	CLOS	FAI	O/C
	2R201TCC0057	A	5R209F05017	D-2	2	A	14	BUTTER	MOTOR	CLOS	FAI	O/C
CC12	CCW Return from RHR Pump and Heat Exchanger - Trains A, B, and C											
	2R201TCC0050	A	5R209F05017	G-4	2	A	16	BUTTER	MOTOR	OPEN	FAI	O/C
	2R201TCC0190	A	5R209F05019	H-4	2	A	16	BUTTER	MOTOR	OPEN	FAI	O/C
	2R201TCC0189	A	5R209F05019	H-4	2	A	16	BUTTER	MOTOR	OPEN	FAI	O/C

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	2R201TCC0049	A	5R209F05017	G-4	2	A	16	BUTTER	MOTOR	OPEN	FAI	O/C
	2R201TCC0129	A	5R209F05018	G-4	2	A	16	BUTTER	MOTOR	OPEN	FAI	O/C
	2R201TCC0130	A	5R209F05018	G-4	2	A	16	BUTTER	MOTOR	OPEN	FAI	O/C
CC13	Chilled Water Return from RCFCs Outside Cntmt. Isolation MOV (Trains A, B, and C)											
	2R201TCC0209	A	5R209F05019	C-4	2	A	8	BUTTER	MOTOR	OPEN	FAI	C
	2R201TCC0149	A	5R209F05018	C-4	2	A	8	BUTTER	MOTOR	OPEN	FAI	C
	2R201TCC0070 .	A	5R209F05017	C-4	2	A	8	BUTTER	MOTOR	OPEN	FAI	C
CC14	Chilled Water Supply to RCFCs Outside Cntmt. Isolation MOV (Trains A, B, and C)											
	2R201TCC0199	A	5R209F05019	D-2	2	A	14	BUTTER	MOTOR	OPEN	FAI	C
	2R201TCC0059	A	5R209F05017	D-2	2	A	14	BUTTER	MOTOR	OPEN	FAI	C
	2R201TCC0137	A	5R209F05018	D-2	2	A	14	BUTTER	MOTOR	OPEN	FAI	C
CC15	CCW Supply Header to Spent Fuel Pool Heat Exchanger, First and Second Isolation											
	3R201TCC0032	A	5R209F05020	E-6	3	B	18	BUTTER	MOTOR	EITH	FAI	C
	3R201TCC0447	A	5R209F05020	E-7	3	B	18	BUTTER	MOTOR	EITH	FAI	C
CC16	CCW Supply Header to Non-Safety Loads, First and Second Isolation											
	3R201TCC0236	A	5R209F05020	D-6	3	B	18	BUTTER	MOTOR	OPEN	N/A	C
	3R201TCC0235	A	5R209F05020	D-7	3	B	18	BUTTER	MOTOR	OPEN	N/A	C
CC17	CCW Supply to Excess Letdown Heat Exchanger Isolation MOV											
	3R201TCC0393	A	5R209F05021	G-3	3	B	4	BUTTER	MOTOR	OPEN	FAI	C
CC18	CCW Supply Header Isolation to Charging Pumps (Trains A, B, and C)											
	3R201TCC0771	A	5R209F05020	G-7	3	B	6	BUTTER	MOTOR	EITH	FAI	O/C
	3R201TCC0768	A	5R209F05020	F-7	3	B	6	BUTTER	MOTOR	EITH	FAI	O/C
	3R201TCC0770	A	5R209F05020	G-7	3	B	6	BUTTER	MOTOR	EITH	FAI	O/C
CC19	CCW Return Isolation from Charging Pumps (Trains A, B, and C)											
	3R201TCC0774	A	5R209F05020	B-7	3	B	6	BUTTER	MOTOR	EITH	FAI	O/C

GROUP	GROUP DESCRIPTION						VALVE DATA			VALVE POSITIONS		
	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	3R201TCC0775	A	5R209F05020	A-7	3	B	6	BUTTER	MOTOR	EITH	FAI	O/C
	3R201TCC0772	A	5R209F05020	B-7	3	B	6	BUTTER	MOTOR	EITH	FAI	O/C
CC20	CCW Supply to RCDT Ht. Exch. and Excess Letdown											
	3R201TCC0297	A	5R209F05021	G-7	3	B	6	BUTTER	MOTOR	EITH	N/A	C
CC21	CCW Supply to RCDT Ht. Exch.											
	3R201TCC0392	A	5R209F05021	G-3	3	B	4	GATE	MOTOR	OPEN	FAI	C
CC22	CCW Supply to RCP Coolers Outside Cntmt Isolation MOVs											
	2R201TCC0291	A	5R209F05021	H-8	2	A	12	BUTTER	MOTOR	OPEN	FAI	C
	2R201TCC0318	A	5R209F05021	H-8	2	A	12	BUTTER	MOTOR	OPEN	FAI	C
CC23	CCW Return from RCP Coolers, Cntmt Isolation MOVs											
	2R201TCC0403	A	5R209F05021	B-1	2	A	12	BUTTER	MOTOR	OPEN	FAI	C
	2R201TCC0404	A	5R209F05021	H-1	2	A	12	BUTTER	MOTOR	OPEN	FAI	C
	2R201TCC0542	A	5R209F05021	B-1	2	A	12	BUTTER	MOTOR	OPEN	FAI	C
CC24	Chilled Water Return for the RCFCs, Outside Cntmt Isolation Valve (Trains A, B, and C)											
	C1CCFV0863	A	5R209F05017	C-4	2	A	8	BUTTER	MOTOR	OPEN	CLOS	C
	B1CCFV0862	A	5R209F05017	B-4	2	A	8	BUTTER	MOTOR	OPEN	CLOS	C
	A1CCFV0864	A	5R209F05017	C-4	2	A	8	BUTTER	MOTOR	OPEN	CLOS	C
CC25	CCW Supply Header to Post Accident Sampling System, First and Second Isolation											
	B1CCFV4541	A	5R209F05020	D-8	3	B	1.5	GATE	SOLENO	CLOS	CLOS	C
	A1CCFV4540	A	5R209F05020	D-7	3	B	1.5	GATE	SOLENO	OPEN	CLOS	C
CC26	CCW Common Return Header to CCW Pump Suction Check Valve (Trains A, B, and C)											
	3R201TCC0131	A	5R209F05020	C-7	3	C	24	CHECK	SELF	OPEN	N/A	O/C
	3R201TCC0051	A	5R209F05020	C-7	3	C	24	CHECK	SELF	OPEN	N/A	O/C
	3R201TCC0191	A	5R209F05020	B-7	3	C	24	CHECK	SELF	OPEN	N/A	O/C
CC27	CCW Pump Discharge Check Valve to Common Supply Header (Trains A, B, and C)											
	3R201TCC0311	A	5R209F05020	E-7	3	C	24	CHECK	SELF	EITH	N/A	O

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	3R201TCC0315	A	5R209F05020	F-7	3	C	24	CHECK	SELF	EITH	N/A	O
	3R201TCC0313	A	5R209F05020	E-7	3	C	24	CHECK	SELF	EITH	N/A	O
CC28	CCW Supply to RCFCs Inside Cntmt Isolation Check Valve (Trains A, B, and C)											
	2R201TCC0198	A	5R209F05019	D-2	2	A/C	14	CHECK	SELF	OPEN	N/A	O/C
	2R201TCC0058	A	5R209F05017	D-2	2	A/C	14	CHECK	SELF	OPEN	N/A	O/C
	2R201TCC0138	A	5R209F05018	D-2	2	A/C	14	CHECK	SELF	OPEN	N/A	O/C
CC29	CCW Supply to RHR Pump and Heat Exchanger Inside Cntmt Isolation Check Valve (Trains A, B, and C)											
	2R201TCC0013	A	5R209F05017	E-2	2	A/C	16	CHECK	SELF	CLOS	N/A	O/C
	2R201TCC0123	A	5R209F05018	E-2	2	A/C	16	CHECK	SELF	CLOS	N/A	O/C
	2R201TCC0183	A	5R209F05019	E-2	2	A/C	16	CHECK	SELF	CLOS	N/A	O/C
CC30	CCW Return for RCDT Heat Exchanger Check Valves											
	3R201TCC0540	A	5R209F05021	D-1	3	C	4	CHECK	SELF	OPEN	N/A	C
	3R201TCC0541	A	5R209F05021	D-1	3	C	4	CHECK	SELF	OPEN	N/A	C
CC31	CCW Return for Excess Letdown Heat Exchanger Check Valves											
	3R201TCC0763	A	5R209F05021	C-2	3	C	6	CHECK	SELF	OPEN	N/A	C
	3R201TCC0402	A	5R209F05021	C-2	3	C	6	CHECK	SELF	OPEN	N/A	C
CC32	CCW Supply to RCPs Inside Containment Isolation Check Valve											
	2R201TCC0319	A	5R209F05021	G-8	2	A/C	12	CHECK	SELF	OPEN	N/A	C
CC33	RCP Thermal Barrier Leak Isolation Valves											
	N1CCFV4620	A	5R209F05021	B-6	3	C	3	GLOBE	SELF	OPEN	OPEN	C
	N1CCFV4627	A	5R209F05021	B-3	3	C	3	GLOBE	SELF	OPEN	OPEN	C
	N1CCFV4626	A	5R209F05021	B-3	3	C	3	GLOBE	SELF	OPEN	OPEN	C
	N1CCFV4621	A	5R209F05021	B-6	3	C	3	GLOBE	SELF	OPEN	OPEN	C
	N1CCFV4633	A	5R209F05021	E-3	3	C	3	GLOBE	SELF	OPEN	OPEN	C
	N1CCFV4638	A	5R209F05021	E-6	3	C	3	GLOBE	SELF	OPEN	OPEN	C

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	N1CCFV4639	A	5R209F05021	E-6	3	C	3	GLOBE	SELF	OPEN	OPEN	C
	N1CCFV4632	A	5R209F05021	E-3	3	C	3	GLOBE	SELF	OPEN	OPEN	C
CC34	Cross Connect Valves for CCW Supply and Return for Charging Pumps											
	A1CCFV4656	A	5R209F05020	G-7	3	B	6	BUTTER	AIR	OPEN	CLOS	C
	A1CCFV4657	A	5R209F05020	A-7	3	B	6	BUTTER	AIR	CLOS	CLOS	C
CC35	CCW Common Return Header Pressure Relief Valve											
	N1CCPSV4492	A	5R209F05020	B7	3	C	1.5	RELIEF	SELF	CLOS	N/A	O
CC36	CCW Heat Exchangers A, B, C Outlet Pressure Relief Valves											
	N1CCPSV4521	A	5R209F05019	B6	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4511	A	5R209F05017	B5	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4516	A	5R209F05018	B6	3	C	1	RELIEF	SELF	CLOS	N/A	O
CC37	RHR Heat Exchanger A, B, C CCW Return Pressure Relief Valves											
	N1CCPSV4566	A	5R209F05019	G2	3	C	1.5	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4532	A	5R209F05017	G2	3	C	1.5	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4549	A	5R209F05018	G2	3	C	1.5	RELIEF	SELF	CLOS	N/A	O
CC38	RHR Pump A, B, C CCW Return Pressure Relief Valves											
	N1CCPSV4533	A	5R209F05017	G3	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4550	A	5R209F05018	G3	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4567	A	5R209F05019	G3	3	C	1	RELIEF	SELF	CLOS	N/A	O
CC39	RCFC 11(21)A, B, C Chilled Water/CCW Return Pressure Relief Valves											
	N1CCPSV4556	A	5R209F05018	C4	3	C	1.5	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4537	A	5R209F05017	E4	3	C	1.5	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4554	A	5R209F05018	E4	3	C	1.5	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4573	A	5R209F05019	C3	3	C	1.5	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4571	A	5R209F05019	E4	3	C	1.5	RELIEF	SELF	CLOS	N/A	O

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	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
CC40	CCP A, B, C Lube Oil and AHU Coolers CCW Return Pressure Relief Valves											
	N1CCPSV4588	A	5R209F05020	G5	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4580	A	5R209F05020	G6	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4582	A	5R209F05020	G4	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4586	A	5R209F05020	G3	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4613	A	5R209F05020	E2	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4584	A	5R209F05020	G5	3	C	1	RELIEF	SELF	CLOS	N/A	O
CC41	RCP A, B, C Upper and Lower Lube Oil Cooler CCW Return Pressure Relief Valves											
	N1CCPSV4616	A	5R209F05021	C6	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4622	A	5R209F05021	C3	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4624	A	5R209F05021	B3	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4628	A	5R209F05021	F3	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4630	A	5R209F05021	E3	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4634	A	5R209F05021	F6	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4636	A	5R209F05021	E6	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4618	A	5R209F05021	B6	3	C	1	RELIEF	SELF	CLOS	N/A	O
CC42	RCP A, B, C, D Thermal Barrier CCW Return Pressure Relief Valves											
	N1CCPSV4638	A	5R209F05021	D6	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4632	A	5R209F05021	D3	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4626	A	5R209F05021	A3	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4620	A	5R209F05021	A6	3	C	1	RELIEF	SELF	CLOS	N/A	O
CC43	RCP and Heat Exchangers CCW Return Header Pressure Relief Valves											
	N1CCPSV4639	A	5R209F05021	C2	3	C	3	RELIEF	SELF	CLOS	N/A	O
CC44	RCP A, B, C, D Upper and Lower Motor Air Cooler CCW Return Pressure Relief Valves											
	N1CCPSV4647A	A	5R209F05021	F3	3	C	1	RELIEF	SELF	CLOS	N/A	O

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	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	N1CCPSV4648	A	5R209F05021	G6	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4647	A	5R209F05021	G3	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4646A	A	5R209F05021	C3	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4646	A	5R209F05021	D3	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4645A	A	5R209F05021	C7	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4645	A	5R209F05021	D7	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CCPSV4648A	A	5R209F05021	F6	3	C	1	RELIEF	SELF	CLOS	N/A	O
CCPP	Component Cooling Water Pumps											
	3R201NPA101A											
CH01	EAB Control Room Envelope Air Handling Unit Outlet Temperature Valve (Trains A, B, and C)											
	A1CHTV9476A	A	3V119V10002	F-7	3	B	2	BUTTER	AIR	THRO	OPEN	O
	A1CHTV9476B	A	3V119V10002	E-7	3	B	2	BUTTER	AIR	THRO	CLOS	C
	B1CHTV9486A	A	3V119V10002	F-4	3	B	2	BUTTER	AIR	THRO	OPEN	O
	B1CHTV9486B	A	3V119V10002	E-4	3	B	2	BUTTER	AIR	THRO	CLOS	C
	C1CHTV9496A	A	3V119V10002	F-1	3	B	2	BUTTER	AIR	THRO	OPEN	O
	C1CHTV9496B	A	3V119V10002	E-1	3	B	2	BUTTER	AIR	THRO	CLOS	C
CH02	EAB Main Supply Air Handling Unit Outlet Temperature Valve (Trains A, B, and C)											
	A1CHTV9477B	A	3V119V10002	C-6	3	B	4	BUTTER	AIR	THRO	CLOS	C
	C1CHTV9497B	A	3V119V10002	C-1	3	B	4	BUTTER	AIR	THRO	CLOS	C
	C1CHTV9497A	A	3V119V10002	C-1	3	B	4	BUTTER	AIR	THRO	OPEN	O
	B1CHTV9487A	A	3V119V10002	C-4	3	B	4	BUTTER	AIR	THRO	OPEN	O
	A1CHTV9477A	A	3V119V10002	C-6	3	B	4	BUTTER	AIR	THRO	OPEN	O
	B1CHTV9487B	A	3V119V10002	C-4	3	B	4	BUTTER	AIR	THRO	CLOS	C
CH05	Train A, B, C Essential Chilled Water Expansion Tank Pressure Relief Valves											

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	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	N1CHPSV9471	A	5V119V10001	H7	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CHPSV9491	A	5V119V10001	C7	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CHPSV9481	A	5V119V10001	E7	3	C	1	RELIEF	SELF	CLOS	N/A	O
CH06	Train A, B, C Essential Chilled Water Expansion Tank Nitrogen Supply Pressure Relief Valves											
	N1CHPSV9481A	A	5V119V10001	E7	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CHPSV9491A	A	5V119V10001	C7	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1CHPSV9471A	A	5V119V10001	H7	3	C	1	RELIEF	SELF	CLOS	N/A	O
CH07	Essential Chilled Water Chiller 11(21) A, B, C Outlet Pressure Relief Valves											
	N1CHPSV9473	A	5V119V10001	G6	3	C	3	RELIEF	SELF	CLOS	N/A	O
	N1CHPSV9493	A	5V119V10001	B6	3	C	3	RELIEF	SELF	CLOS	N/A	O
	N1CHPSV9483	A	5V119V10001	E6	3	C	3	RELIEF	SELF	CLOS	N/A	O
CH08	Essential Chilled Water Chiller 12(22) A, B, C Outlet Pressure Relief Valves											
	N1CHPSV9514	A	5V119V10001	B4	3	C	4	RELIEF	SELF	CLOS	N/A	O
	N1CHPSV9508	A	5V119V10001	E4	3	C	4	RELIEF	SELF	CLOS	N/A	O
	N1CHPSV9502	A	5V119V10001	G4	3	C	4	RELIEF	SELF	CLOS	N/A	O
CM01	RCB Air Sample Select Valves for Cntmt Hydrogen Monitoring System											
	A1CMFV4124	A	5Z169Z00046	F-6	2	B	1	GATE	SOLENO	CLOS	CLOS	O
	A1CMFV4100	A	5Z169Z00046	G-6	2	B	1	GATE	SOLENO	EITH	CLOS	O
	A1CMFV4125	A	5Z169Z00046	F-6	2	B	1	GATE	SOLENO	CLOS	CLOS	O
	A1CMFV4126	A	5Z169Z00046	E-6	2	B	1	GATE	SOLENO	CLOS	CLOS	O
	C1CMFV4131	A	5Z169Z00046	C-6	2	B	1	GATE	SOLENO	CLOS	CLOS	O
	C1CMFV4103	A	5Z169Z00046	E-6	2	B	1	GATE	SOLENO	CLOS	CLOS	O
	C1CMFV4130	A	5Z169Z00046	D-6	2	B	1	GATE	SOLENO	CLOS	CLOS	O
	C1CMFV4129	A	5Z169Z00046	D-6	2	B	1	GATE	SOLENO	CLOS	CLOS	O
CM02	Cntmt Hydrogen Monitoring System Inside and Outside CIVs											

GROUP	GROUP DESCRIPTION TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	VALVE DATA Size	Type	Actuator	VALVE POSITIONS Normal	Failsafe	Safety Func.
	A1CMFV4101	A	5Z169Z00046	F-4	2	A	1	GATE	SOLENO	CLOS	CLOS	O/C
	C1CMFV4134	A	5Z169Z00046	C-5	2	A	1	GATE	SOLENO	CLOS	CLOS	O/C
	A1CMFV4128	A	5Z169Z00046	E-5	2	A	1	GATE	SOLENO	CLOS	CLOS	O/C
	A1CMFV4135	A	5Z169Z00046	F-5	2	A	1	GATE	SOLENO	CLOS	CLOS	O/C
	C1CMFV4136	A	5Z169Z00046	D-5	2	A	1	GATE	SOLENO	CLOS	CLOS	O/C
	C1CMFV4133	A	5Z169Z00046	C-4	2	A	1	GATE	SOLENO	CLOS	CLOS	O/C
	A1CMFV4127	A	5Z169Z00046	E-4	2	A	1	GATE	SOLENO	CLOS	CLOS	O/C
	C1CMFV4104	A	5Z169Z00046	D-4	2	A	1	GATE	SOLENO	CLOS	CLOS	O/C
CS01	Containment Spray Pump Discharge Outside Cntmt Isolation MOVs											
	2N101XCS0001B	A	5N109F05037	E-6	2	A	8	GATE	MOTOR	CLOS	FAI	O/C
	2N101XCS0001C	A	5N109F05037	C-6	2	A	8	GATE	MOTOR	CLOS	FAI	O/C
	2N101XCS0001A	A	5N109F05037	G-6	2	A	8	GATE	MOTOR	CLOS	FAI	O/C
CS02	Containment Spray Header Inside Cntmt Isolation Check Valves											
	2N101XCS0004	A	5N109F05037	E-8	2	A/C	8	CHECK	SELF	CLOS	N/A	O/C
	2N101XCS0002	A	5N109F05037	G-7	2	A/C	8	CHECK	SELF	CLOS	N/A	O/C
	2N101XCS0005	A	5N109F05037	D-8	2	A/C	8	CHECK	SELF	CLOS	N/A	O/C
	2N101XCS0006	A	5N109F05037	C-7	2	A/C	8	CHECK	SELF	CLOS	N/A	O/C
CV01	Reactor Coolant Auxiliary Spray Valve											
	N1CVLV3119	A	5R179F05	F-7	1	B	2	GLOBE	AIR	CLOS	CLOS	O
CV02	Centrifugal Charging Pump Minimum Recirc. Control Valves											
	N1CVFCV0201	A	5R179F05007	C-6	2	B	2	GLOBE	AIR	EITH	OPEN	O
	N1CVFCV0202	A	5R179F05007	D-6	2	B	2	GLOBE	AIR	EITH	OPEN	O
CV03	RCS Letdown Line Inside Cntmt Isolation Bypass Check Valve (CV0022)											
	2R171TCV0022	A	5R179F05005	H-3	2	A/C	0.75	CHECK	SELF	CLOS	N/A	O/C
CV04	RCS Seal Water Return Inside Cntmt Isolation Bypass Check Valve (CV0078)											

GROUP	GROUP DESCRIPTION						VALVE DATA			VALVE POSITIONS		
	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	2R171TCV0078	A	5R179F05005	F-3	2	A/C	0.75	CHECK	SELF	CLOS	N/A	O/C
CV05	(CV0346,351) BAT Pump recirc valves											
	3R171TCV0351	A	5R179F05009	E-6	3	C	0.75	CHECK	SELF	EITH	N/A	O
	3R171TCV0346	A	5R179F05009	D-5	3	C	0.75	CHECK	SELF	EITH	N/A	O
CV06	RCP Seal Injection Check Valve (Class 1 Boundary Isolation)											
	1R171TCV0037D	A	5R179F05005	C-7	1	C	2	CHECK	SELF	OPEN	N/A	O/C
	1R171TCV0036C	A	5R179F05005	C-7	1	C	2	CHECK	SELF	OPEN	N/A	O/C
	1R171TCV0037A	A	5R179F05005	C-7	1	C	2	CHECK	SELF	OPEN	N/A	O/C
	1R171TCV0037B	A	5R179F05005	C-7	1	C	2	CHECK	SELF	OPEN	N/A	O/C
	1R171TCV0037C	A	5R179F05005	C-7	1	C	2	CHECK	SELF	OPEN	N/A	O/C
	1R171TCV0036A	A	5R179F05005	C-7	1	C	2	CHECK	SELF	OPEN	N/A	O/C
	1R171TCV0036B	A	5R179F05005	C-7	1	C	2	CHECK	SELF	OPEN	N/A	O/C
	1R171TCV0036D	A	5R179F05005	C-7	1	C	2	CHECK	SELF	OPEN	N/A	O/C
CV07	Seal Injection to RCPs Inside Cntmt Isolation Check Valves											
	2R171TCV0034A	A	5R179F05005	C-8	2	A/C	2	CHECK	SELF	OPEN	N/A	O/C
	2R171TCV0034D	A	5R179F05005	C-8	2	A/C	2	CHECK	SELF	OPEN	N/A	O/C
	2R171TCV0034C	A	5R179F05005	C-8	2	A/C	2	CHECK	SELF	OPEN	N/A	O/C
	2R171TCV0034B	A	5R179F05005	C-8	2	A/C	2	CHECK	SELF	OPEN	N/A	O/C
CV08	Boric Acid Polishing Return to Boric Acid Tank											
	3R171TCV0636	A	5R179F05009	E-5	3	C	2	CHECK	SELF	OPEN	N/A	C
	3R171TCV0637	A	5R179F05009	F-5	3	C	2	CHECK	SELF	OPEN	N/A	C
	3R171TCV0638	A	5R179F05009	F-6	3	C	2	CHECK	SELF	OPEN	N/A	C
	3R171TCV0635	A	5R179F05009	E-5	3	C	2	CHECK	SELF	OPEN	N/A	C
CV09	Centrifugal Charging Pump Minimum Recirc. Check Valves											
	2R171TCV0234A	A	5R179F05007	B-6	2	C	3	CHECK	SELF	EITH	N/A	O

GROUP	GROUP DESCRIPTION						VALVE DATA			VALVE POSITIONS		
	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	2R171TCV0234B	A	5R179F05007	D-6	2	C	3	CHECK	SELF	EITH	N/A	O
CV10	Reactor Coolant Auxiliary Spray Inlet Check Valve (CV0009)											
	1R171TCV0009	A	5R179F05005	F-8	1	C	2	CHECK	SA	CLOS	N/A	O
CV11	CVCS SEAL WATER INJECTION FLOW CONTROL VALVE											
	C1CVHCV0218	A	5R179F05007	B-7	2	B	2	GLOBE	AIR	CLOS	OPEN	O
CV12	Letdown Orifice Header Isolation Valve											
	C1CVFV0011	A	5R179F05005	G-6	2	B	3	GLOBE	AIR	OPEN	CLOS	C
CV13	RCS Charging Flow Control Valve											
	A1CVFCV0205	A	5R179F05009	E-7	2	B	3	GLOBE	AIR	EITH	OPEN	O
CV14	Manual Alternate Borate Check Valve											
	2R171XCV0639	A	5R179F05007	E-4	2	C	2	CHECK	SELF	CLOS	N/A	O/C
CV15	Charging Header Check Valve (CV671)											
	2R171XCV0671	A	5R179F05007	B-6	2	C	2	CHECK	SELF	CLOS	N/A	O/C
CV16	Boric Acid Supply to Concentrated BA Polishing Demineralizer Isolation Valves											
	A1CVFV8400A	A	5R179F05009	D-8	3	B	2	DIAPHR	AIR	OPEN	CLOS	C
	B1CVFV8400B	A	5R179F05009	C-8	3	B	2	DIAPHR	AIR	OPEN	CLOS	C
CV19	RCS Charging Outside Cntmt Isolation MOV											
	2R171XCV0025	A	5R179F05005	G-3	2	A	4	GATE	MOTOR	OPEN	FAI	O/C
CV20	RCS Letdown Isolation (Class 1 Boundary Isolation)											
	1R171XCV0468	A	5R179F05005	G-7	1	B	4	GATE	MOTOR	OPEN	FAI	C
	1R171XCV0465	A	5R179F05005	G-8	1	B	4	GATE	MOTOR	OPEN	FAI	C
CV21	Centrifugal Charging Pump Discharge Isolation MOVs											
	2R171XCV8377A	A	5R179F05007	B-6	2	B	3	GATE	MOTOR	OPEN	FAI	O/C
	2R171XCV8377B	A	5R179F05007	D-6	2	B	3	GATE	MOTOR	OPEN	FAI	O/C
CV22	Volume Control Tank Outlet Isolation MOVs											
	2R171XCV0112B	A	5R179F05007	E-4	2	B	6	GATE	MOTOR	EITH	FAI	
	2R171XCV0113A	A	5R179F05007	E-4	2	B	6	GATE	MOTOR	EITH	FAI	

GROUP	GROUP DESCRIPTION / TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
CV23	Reactor Water Storage Tank to Charging Pump Suction Header Isolation MOVs											
	2R171XCV0113B	A	5R179F05007	C-4	2	B	6	GATE	MOTOR	EITH	FAI	
	2R171XCV0112C	A	5R179F05007	C-4	2	B	6	GATE	MOTOR	EITH	FAI	
CV24	Alternate Boric Acid Make-Up Supply Isolation MOV (CV0218)											
	2R171XCV0218	A	5R179F05007	B-3	2	B	4	GATE	MOTOR	CLOS	FAI	O
CV25	RCS Normal and Alternate Charging Flow Isolation MOVs											
	2R171XCV0003	A	5R179F05005	G-7	2	B	4	GATE	MOTOR	EITH	FAI	O/C
	2R171XCV0006	A	5R179F05005	F-7	2	B	4	GATE	MOTOR	CLOS	FAI	O/C
CV26	RCS Letdown Inside and Outside Cntmt Isolation MOVs											
	2R171XCV0023	A	5R179F05005	H-3	2	A	4	GATE	MOTOR	OPEN	FAI	C
	2R171XCV0024	A	5R179F05005	H-3	2	A	4	GATE	MOTOR	OPEN	FAI	C
CV27	RCP Seal Injection Outside Cntmt Isolation MOVs											
	2R171TCV0033A	A	5R179F05005	C-8	2	A	2	DIAPHR	MOTOR	OPEN	FAI	O/C
	2R171TCV0033B	A	5R179F05005	C-8	2	A	2	DIAPHR	MOTOR	OPEN	FAI	O/C
	2R171TCV0033C	A	5R179F05005	C-8	2	A	2	DIAPHR	MOTOR	OPEN	FAI	O/C
	2R171TCV0033D	A	5R179F05005	C-8	2	A	2	DIAPHR	MOTOR	OPEN	FAI	O/C
CV29	RCP Seal Water Return Inside and Outside Cntmt Isolation MOVs											
	2R171TCV0079	A	5R179F05005	E-3	2	A	2	DIAPHR	MOTOR	OPEN	FAI	C
CV30	RCS Excess Letdown Heat Exchanger Inlet Isolation MOVs (Class 1 Boundary Isolation)											
	1R171TCV0083	A	5R179F05005	F-5	1	B	1	DIAPHR	MOTOR	EITH	FAI	C
	1R171TCV0082	A	5R179F05005	F-5	1	B	1	DIAPHR	MOTOR	EITH	FAI	C
CV31	CVCS Alternate Immediate Boration Isolation Valve (CV0221)											
	2R171TCV0221	A	5R179F05007	E-4	2	B	2	DIAPHR	MANUAL	CLOS	N/A	
CV32	Charging Pump B Discharge Bypass Control Valve											
	A1CVHCV0206	A	5R179F05007	D-6	2	B	1	GLOBE	SOLENO	CLOS	CLOS	O
CV33	Centrifugal Charging Pump Discharge Check Valves											

GROUP	GROUP DESCRIPTION						VALVE DATA			VALVE POSITIONS		
	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	2R171XCV0235A	A	5R179F05007	B-6	2	C	3	CHECK	SELF	EITH	N/A	O/C
	2R171XCV0235B	A	5R179F05007	D-6	2	C	3	CHECK	SELF	EITH	N/A	O/C
CV34	(CV0334) check valve											
	3R171XCV0334	A	5R179F05009	E-4	2	C	3	CHECK	SELF	CLOS	N/A	O
CV35	RC Filters out to RHR Outside Cntmt Isolation Manual Valve											
	2R171XCV0157	P	5R179F05006	F-2	2	A	4	GATE	MANUAL	CLOS	N/A	C
CV37	Charging Header Check Valve											
	2R171XCV0670	A	5R179F05007	D-6	2	C	2	CHECK	SELF	CLOS	N/A	O/C
CV38	RCS Normal and Alternate Charging Check Valves (Class 1 Boundary Valves)											
	1R171XCV0002	A	5R179F05005	G-8	1	C	4	CHECK	SELF	EITH	N/A	O/C
	1R171XCV0004	A	5R179F05005	F-8	1	C	4	CHECK	SELF	EITH	N/A	O/C
	1R171XCV0001	A	5R179F05005	G-8	1	C	4	CHECK	SELF	EITH	N/A	O/C
	1R171XCV0005	A	5R179F05005	F-8	1	C	4	CHECK	SELF	EITH	N/A	O/C
CV40	RCS Charging Inside Cntmt Isolation Check Valve.											
	2R171XCV0026	A	5R179F05005	G-3	2	A/C	4	CHECK	SELF	OPEN	N/A	O/C
CV41	Alternate Boric Acid Make-Up Supply Isolation Check Valve (CV0217)											
	2R171XCV0217	A	5R179F05007	B-3	2	C	4	CHECK	SELF	CLOS	N/A	O
CV42	Boric Acid Pump Discharge Check Valves (CV349,338)											
	3R171XCV0338	A	5R179F05009	D-6	3	C	4	CHECK	SELF	EITH	N/A	O/C
	3R171XCV0349	A	5R179F05009	C-6	3	C	4	CHECK	SELF	EITH	N/A	O/C
CV43	RC Filters out to RHR Inside Cntmt Isolation Check Valve											
	2R171XCV0158	P	5R179F05006	F-2	2	A/C	4	CHECK	SELF	CLOS	N/A	C
CV44	Reactor Water Storage Tank to Charging Pump Suction Header Isolation Check Valve											
	2R171XCV0224	A	5R179F05007	B-4	2	C	6	CHECK	SELF	EITH	N/A	O
DW01	Demineralizer Water to the RCB Inside Cntmt Isolation Check Valve											
	2S191TDW0502	P	5S199F05034	F-3	2	A/C	4	CHECK	SELF	CLOS	N/A	C
DW02	Demineralizer Water to the RCB Outside Cntmt Isolation Manual Valve											

GROUP	GROUP DESCRIPTION TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	VALVE DATA Size	Type	Actuator	VALVE POSITIONS Normal	Failsafe	Safety Func.
	2S191TDW0501	P	5S199F05034	F-4	2	A	4	DIAPHR	MANUAL	CLOS	N/A	C
ED01	Containment Normal Sump Discharge Outside Cntmt Isolation Valve (FV7800)											
	A1EDFV7800	A	5Q069F05030	G-7	2	A	3	GLOBE	AIR	O/C	CLOS	C
ED02	Containment Normal Sump Discharge Inside Cntmt Isolation MOV (ED0064)											
	2Q061TED0064	A	5Q069F05030	G-7	2	A	3	GLOBE	MOTOR	O/C	FAI	C
EW01	Essential Cooling Water Blowdown Isolation Valve (Trains A, B, and C)											
	B1EWFV6936	A	5R289F05038	E-5	3	B	4	GLOBE	AIR	OPEN	CLOS	C
	A1EWFV6935	A	5R289F05038	E-5	3	B	4	GLOBE	AIR	OPEN	CLOS	C
	C1EWFV6937	A	5R289F05038	E-5	3	B	4	GLOBE	AIR	OPEN	CLOS	C
EW02	Essential Cooling Water Pump Discharge Vent Check Valve (Trains A, B, and C)											
	3R281TEW0370A	A	5R289F05038	C-3	3	C	3	CHECK	SELF	OPEN	N/A	O/C
	3R281TEW0370B	A	5R289F05038	C-3	3	C	3	CHECK	SELF	OPEN	N/A	O/C
	3R281TEW0370C	A	5R289F05038	C-3	3	C	3	CHECK	SELF	OPEN	N/A	O/C
EW03	ECW Screen Wash Booster Pump Discharge Check Valve (Trains A, B, and C)											
	3R281TEW0253	A	5R289F05039	D-7	3	C	3	CHECK	SELF	EITH	N/A	O
	3R281TEW0254	A	5R289F05039	D-5	3	C	3	CHECK	SELF	EITH	N/A	O
	3R281TEW0255	A	5R289F05039	D-2	3	C	3	CHECK	SELF	EITH	N/A	O
EW04	Essential Cooling Water Pump Discharge Strainer Emergency Backflush Check Valve (Trains A, B, and C)											
	3R281TEW0403	A	5R289F05038	C-3	3	C	6	CHECK	SELF	OPEN	N/A	O/C
	3R281TEW0404	A	5R289F05038	C-3	3	C	6	CHECK	SELF	OPEN	N/A	O/C
	3R281TEW0405	A	5R289F05038	C-3	3	C	6	CHECK	SELF	OPEN	N/A	O/C
EW05	Essential Cooling Water Pump Discharge MOV (Trains A, B, and C)											
	3R281TEW0151	A	5R289F05038	C-2	3	B	30	BUTTER	MOTOR	EITH	FAI	O
	3R281TEW0137	A	5R289F05038	C-2	3	B	30	BUTTER	MOTOR	EITH	FAI	O
	3R281TEW0151	A	5R289F05038	C-2	3	B	30	BUTTER	MOTOR	EITH	FAI	O

GROUP	GROUP DESCRIPTION						VALVE DATA			VALVE POSITIONS		
	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	3R281TEW0137	A	5R289F05038	C-2	3	B	30	BUTTER	MOTOR	EITH	FAI	O
	3R281TEW0121	A	5R289F05038	C-2	3	B	30	BUTTER	MOTOR	EITH	FAI	O
	3R281TEW0121	A	5R289F05038	C-2	3	B	30	BUTTER	MOTOR	EITH	FAI	O
EW06	ECW Self-Cleaning Strainer Backflush Throttle Valve (Manual)											
	3R281TEW0188	A	5R289F05038	C-2	3	B	6	BUTTER	MANUAL	OPEN	N/A	O/C
	3R281TEW0189	A	5R289F05038	C-2	3	B	6	BUTTER	MANUAL	OPEN	N/A	O/C
	3R281TEW0190	A	5R289F05038	C-2	3	B	6	BUTTER	MANUAL	OPEN	N/A	O/C
EW07	ECW Self-Cleaning Strainer Emergency Backflush Manual Valve											
	3R281TEW0277	A	5R289F05038	C-2	3	B	6	BUTTER	MANUAL	CLOS	N/A	O/C
	3R281TEW0278	A	5R289F05038	C-2	3	B	6	BUTTER	MANUAL	CLOS	N/A	O/C
	3R281TEW0279	A	5R289F05038	C-2	3	B	6	BUTTER	MANUAL	CLOS	N/A	O/C
EW08	Essential Cooling Water Pump Discharge Check Valve (Trains A, B, and C)											
	3R281TEW0042	A	5R289F05038	C-3	3	C	30	CHECK	SELF	EITH	N/A	O
	3R281TEW0006	A	5R289F05038	C-3	3	C	30	CHECK	SELF	EITH	N/A	O
	3R281TEW0079	A	5R289F05038	C-3	3	C	30	CHECK	SELF	EITH	N/A	O
EW09	ECW Screen Wash Pump Discharge Valve (Trains A, B, and C)											
	A1EWFV6914	A	5R289F05039	D-7	3	B	3	GLOBE	AIR	EITH	OPEN	O
	C1EWFV6934	A	5R289F05039	D-3	3	B	3	GLOBE	AIR	EITH	OPEN	O
	B1EWFV6924	A	5R289F05039	D-5	3	B	3	GLOBE	AIR	EITH	OPEN	O
EW10	CCW Heat Exchanger A, B, C ECW Return Relief Valves											
	N1EWPSV6863	A	5R289F05038	G7	3	C	1.5	RELIEF	SELF	CLOS	N/A	O
	N1EWPSV6873	A	5R289F05038	G7	3	C	1.5	RELIEF	SELF	CLOS	N/A	O
	N1EWPSV6853	A	5R289F05038	G7	3	C	1.5	RELIEF	SELF	CLOS	N/A	O
EW11	Ess. Chlr 11(21)A,B,C/DG11(21),12(22),13(23)/CCW Pump Sup. Clr A,B,C ECW Return Relief Valves											
	N1EWPSV6876	A	5R289F05038	G8	3	C	1	RELIEF	SELF	CLOS	N/A	O

GROUP	GROUP DESCRIPTION TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	VALVE DATA Size	Type	Actuator	VALVE POSITIONS Normal	Failsafe	Safety Func.
	N1EWPSV6874	A	5R289F05038	G5	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1EWPSV6875	A	5R289F05038	G2	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1EWPSV6854	A	5R289F05038	G5	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1EWPSV6856	A	5R289F05038	G8	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1EWPSV6856	A	5R289F05038	G8	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1EWPSV6864	A	5R289F05038	G5	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1EWPSV6865	A	5R289F05038	G2	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1EWPSV6855	A	5R289F05038	G2	3	C	1	RELIEF	SELF	CLOS	N/A	O
EW12	Essential Chiler 12(22) A, B, C ECW Return Relief Valves											
	N1EWPSV6904	A	5R289F05038	G4	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1EWPSV6906	A	5R289F05038	G4	3	C	1	RELIEF	SELF	CLOS	N/A	O
	N1EWPSV6905	A	5R289F05038	G4	3	C	1	RELIEF	SELF	CLOS	N/A	O
EWPP	EW Pumps											
	3R281NPA101A											
FC01	SFP Pump Discharge Reactor Cavity ICIV (Manual)											
	2R211XFC0050	P	5R219F05028	B-6	2	A	3	GATE	MANUAL	CLOS	N/A	C
FC02	SFP Pump Cooling Supply and Return from In-Cntmt Storage Area CIV (Manual)											
	2R211XFC0013E	P	5R219F05028	B-6	2	A	10	GATE	MANUAL	CLOS	N/A	C
	2R211XFC0007C	P	5R219F05028	B-4	2	A	10	GATE	MANUAL	CLOS	N/A	C
	2R211XFC0013F	P	5R219F05028	B-6	2	A	10	GATE	MANUAL	CLOS	N/A	C
	2R211XFC0006C	P	5R219F05028	B-5	2	A	10	GATE	MANUAL	CLOS	N/A	C
FP01	Fire Protection to the RCB Inside Cntmt Isolation Check Valve											
	2Q271TFP0943	A	5Q279F05047	E-8	2	A/C	6	CHECK	SELF	CLOS	N/A	C
FP02	Fire Protection to the RCB Outside Cntmt Isolation MOV											
	2Q271TFP0756	A	5Q279F05047	E-8	2	A	6	GATE	MOTOR	CLOS	FAI	C

GROUP	GROUP DESCRIPTION						VALVE DATA			VALVE POSITIONS		
	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
FW01	Feedwater to the Steam Generator Isolation Valves											
	A1FWFV7144	A	5S139F00063	G-2	2	B	18	GATE	HYDRAU	OPEN	CLOS	C
	A1FWFV7141	A	5S139F00063	G-8	2	B	18	GATE	HYDRAU	OPEN	CLOS	C
	A1FWFV7142	A	5S139F00063	G-6	2	B	18	GATE	HYDRAU	OPEN	CLOS	C
	A1FWFV7143	A	5S139F00063	G-4	2	B	18	GATE	HYDRAU	OPEN	CLOS	C
FW02	Feedwater flow control valves											
	N1FWFCV0553	A	5S139F00063	D-4	NNS	B	16	ANGLE	AIR	OPEN	CLOS	C
	N1FWFCV0552	A	5S139F00063	D-6	NNS	B	16	ANGLE	AIR	OPEN	CLOS	C
	N1FWFCV0554	A	5S139F00063	D-2	NNS	B	16	ANGLE	AIR	OPEN	CLOS	C
	N1FWFCV0551	A	5S139F00063	D-8	NNS	B	16	ANGLE	AIR	OPEN	CLOS	C
FW03	Feedwater Bypass Flow Control Valves											
	N1FWFV7152	A	5S139F00063	D-5	NNS	B	4	GLOBE	AIR	CLOS	CLOS	C
	N1FWFV7153	A	5S139F00063	D-3	NNS	B	4	GLOBE	AIR	CLOS	CLOS	C
	N1FWFV7151	A	5S139F00063	D-7	NNS	B	4	GLOBE	AIR	CLOS	CLOS	C
	N1FWFV7154	A	5S139F00063	D-1	NNS	B	4	GLOBE	AIR	CLOS	CLOS	C
FW04	Steam Generator Feedwater Inlet Isolation Bypass Valves											
	A1FWFV7148A	P	5S139F00063	G-7	2	B	2	GLOBE	AIR	CLOS	CLOS	C
	B1FWFV7145A	P	5S139F00063	G-1	2	B	2	GLOBE	AIR	CLOS	CLOS	C
	A1FWFV7147A	P	5S139F00063	G-5	2	B	2	GLOBE	AIR	CLOS	CLOS	C
	B1FWFV7146A	P	5S139F00063	G-3	2	B	2	GLOBE	AIR	CLOS	CLOS	C
FW05	Steam Generator Preheater Bypass Valves											
	A1FWFV7192	A	5S139F00063	E-2	2	B	3	GLOBE	AIR	CLOS	CLOS	C
	A1FWFV7191	A	5S139F00063	E-4	2	B	3	GLOBE	AIR	CLOS	CLOS	C
	A1FWFV7190	A	5S139F00063	E-6	2	B	3	GLOBE	AIR	CLOS	CLOS	C
	A1FWFV7189	A	5S139F00063	E-8	2	B	3	GLOBE	AIR	CLOS	CLOS	C

GROUP	GROUP DESCRIPTION						VALVE DATA			VALVE POSITIONS		
	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
HC01	RCB Supplemental Purge Supply and Return Inside Cntmt Isolation MOVs											
	2V141THC0005	A	5V149V00019	B-7	2	A	18	BUTTER	MOTOR	OPEN	FAI	C
	2V141THC0003	A	5V149V00019	F-3	2	A	18	BUTTER	MOTOR	OPEN	FAI	C
HC02	RCB Supplemental Purge Supply and Return Outside Cntmt Isolation AOVs											
	A1HCFV9777	A	5V149V00019	B-6	2	A	18	BUTTER	AIR	OPEN	CLOS	C
	A1HCFV9776	A	5V149V00019	F-4	2	A	18	BUTTER	AIR	OPEN	CLOS	C
HC03	RCB Normal Purge Supply and Exhaust Cntmt Isolation (48") MOVs											
	2V141ZHC0007	A	5V149V00018	G-3	2	A	48	BUTTER	MOTOR	CLOS	FAI	C
	2V141ZHC0010	A	5V149V00018	B-6	2	A	48	BUTTER	MOTOR	CLOS	FAI	C
	2V141ZHC0008	A	5V149V00018	G-2	2	A	48	BUTTER	MOTOR	CLOS	FAI	C
	2V141ZHC0009	A	5V149V00018	B-7	2	A	48	BUTTER	MOTOR	CLOS	FAI	C
IA01	Instrument Air to RCB Inside Cntmt Isolation Check Valve (IA0541)											
	2Q111TIA0541	A	5N109F05040	D-4	2	A/C	2	CHECK	SELF	OPEN	N/A	C
IA02	Instrument Air to RCB Outside Cntmt Isolation Valve (IA8565)											
	B1IAFV8565	A	5N109F05040	D-4	2	A	2	BALL	AIR	OPEN	CLOS	C
MS01	Main Steam Isolation Valves											
	A1MSFSV7414	A	5S109F00016	G-4	2	B	30	GATE	AIR	OPEN	CLOS	C
	A1MSFSV7424	A	5S109F00016	F-4	2	B	30	GATE	AIR	OPEN	CLOS	C
	A1MSFSV7434	A	5S109F00016	D-4	2	B	30	GATE	AIR	OPEN	CLOS	C
	A1MSFSV7444	A	5S109F00016	C-4	2	B	30	GATE	AIR	OPEN	CLOS	C
MS03	Main Steam Power Operated Relief Valves											
	A1MSPV7411	A	5S109F00016	H-6	2	B	8	GLOBE	HYDRAU	CLOS	CLOS	O/C
	B1MSPV7421	A	5S109F00016	F-6	2	B	8	GLOBE	HYDRAU	CLOS	CLOS	O/C
	C1MSPV7431	A	5S109F00016	E-6	2	B	8	GLOBE	HYDRAU	CLOS	CLOS	O/C
	D1MSPV7441	A	5S109F00016	C-6	2	B	8	GLOBE	HYDRAU	CLOS	CLOS	O/C
MS04	Main Steam Bypass Isolation Valves											

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	A1MSFV7422	A	5S109F00016	F-4	2	B	4	GLOBE	AIR	CLOS	CLOS	C
	A1MSFV7432	A	5S109F00016	D-4	2	B	4	GLOBE	AIR	CLOS	CLOS	C
	A1MSFV7442	A	5S109F00016	C-4	2	B	4	GLOBE	AIR	CLOS	CLOS	C
	A1MSFV7412	A	5S109F00016	G-4	2	B	4	GLOBE	AIR	CLOS	CLOS	C
PO01	RCP Motor Oil Return system											
	2R371TPO0217	P	5R149F05042	B-4	2	A	2	DIAPHR	MANUAL	CLOS	N/A	C
	2R371TPO0218	P	5R149F05042	B-3	2	A	2	DIAPHR	MANUAL	CLOS	N/A	C
PS01	Pressurizer Vapor Space Sample Inside Cntmt Isolation Valve (4450)											
	B1PSFV4450	A	5Z329Z00045	H-8	2	A	1	GATE	SOLENO	CLOS	CLOS	C
PS02	RCS Pressurizer and Hot Leg Sample ICIVs											
	C1PSFV4455	A	5Z329Z00045	E-8	2	A	1	GATE	SOLENO	CLOS	CLOS	C
	C1PSFV4454	A	5Z329Z00045	F-8	2	A	1	GATE	SOLENO	CLOS	CLOS	C
	B1PSFV4451	A	5Z329Z00045	G-8	2	A	1	GATE	SOLENO	CLOS	CLOS	C
PS03	RHR and Accumulator Sample ICIVs											
	C1PSFV4824	A	5Z329Z00045	B-8	2	A	1	GATE	SOLENO	CLOS	CLOS	C
	B1PSFV4823	A	5Z329Z00045	D-8	2	A	1	GATE	SOLENO	CLOS	CLOS	C
PS04	Pressurizer Liquid Sample OCIV											
	C1PSFV4451B	A	5Z329Z00045	F-7	2	A	1	GLOBE	AIR	CLOS	CLOS	C
PS05	Pressurizer Vapor Space Sample OCIV											
	C1PSFV4452	A	5Z329Z00045	G-7	2	A	1	GLOBE	AIR	CLOS	CLOS	C
PS07	Primary sampling OCIVs (FV4461 and FV4466, FV 4456)											
	B1PSFV4466	A	5Z329Z00045	B-7	2	A	1	GLOBE	AIR	CLOS	CLOS	C
	C1PSFV4461	A	5Z329Z00045	D-7	2	A	1	GLOBE	AIR	CLOS	CLOS	C
RA01	RCB Atmosphere Rad Monitor Inside and Outside Cntmt Isolation Valves											
	2V141TRA0001	A	5V14900017	G-4	2	A	1	BALL	MOTOR	OPEN	FAI	C
	2V141TRA0004	A	5V14900017	G-4	2	A	1	BALL	MOTOR	OPEN	FAI	C

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	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	2V141TRA0006	A	5V14900017	F-3	2	A	1	BALL	MOTOR	OPEN	FAI	C
	2V141TRA0003	A	5V14900017	F-4	2	A	1	BALL	MOTOR	OPEN	FAI	C
RC01	Pressurizer Relief Tank Vent to Gaseous Waste Processing System Outside Cntmt Isolation Valve (3652)											
	B1RCFV3652	A	5R149F05004	F-4	2	A	1	BALL	AIR	CLOS	CLOS	C
RC02	Reactor Make-up Water to RCP Standpipe and PRT OCIV (3651)											
	B1RCFV3651	A	5R149F05004	E-2	2	A	3	BALL	AIR	OPEN	CLOS	C
RC03	RCS Pressurizer Safety Valves											
	N1RCPSV3451	A	5R149F05003	F-6	1	C	6	RELIEF	SELF	CLOS	N/A	O
	N1RCPSV3452	A	5R149F05003	F-4	1	C	6	RELIEF	SELF	CLOS	N/A	O
	N1RCPSV3450	A	5R149F05003	F-7	1	C	6	RELIEF	SELF	CLOS	N/A	O
RC04	RCS Power Operated Relief Valves											
	B1RCPCV0656A	A	5R149F05003	E-8	1	B	3	GLOBE	SOLENO	CLOS	CLOS	O/C
	A1RCPCV0655A	A	5R149F05003	D-8	1	B	3	GLOBE	SOLENO	CLOS	CLOS	O/C
RC05	RCS PORV Block Valves											
	1R141XRC0001B	A	5R149F05003	E-8	1	B	3	GATE	MOTOR	OPEN	FAI	C
	1R141XRC0001A	A	5R149F05003	E-7	1	B	3	GATE	MOTOR	OPEN	FAI	C
RC06	Reactor Vessel Head Vent Isolation Valves											
	A1RCHV3658A	A	5R149F05001	E-3	2	B	1	GLOBE	SOLENO	CLOS	CLOS	C
	B1RCHV3657B	A	5R149F05001	E-4	1	B	1	GLOBE	SOLENO	CLOS	CLOS	C
	A1RCHV3657A	A	5R149F05001	E-4	2	B	1	GLOBE	SOLENO	CLOS	CLOS	C
	B1RCHV3658B	A	5R149F05001	E-3	1	B	1	GLOBE	SOLENO	CLOS	CLOS	C
RC07	Reactor Vessel Head Vent Throttle Valves											
	B1RCHCV0602	A	5R149F05001	D-2	2	B	1	GLOBE	SOLENO	CLOS	CLOS	O/C
	A1RCHCV0601	A	5R149F05001	E-2	2	B	1	GLOBE	SOLENO	CLOS	CLOS	O/C
RC08	Pressurizer Relief Tank Vent to Gaseous Waste Processing System Inside Cntmt Isolation Valve (3652)											
	A1RCFV3653	A	5R149F05004	F-4	2	A	3	GATE	SOLENO	CLOS	CLOS	C

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	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
RC09	Reactor Make-up Water to RCP Standpipe and PRT Outside Containment Check Valve.											
	2R141XRC0046	A	5R149F05004	E-4	2	A/C	3	CHECK	SELF	OPEN	N/A	C
RD01	RCS Vacuum Degassing from RCB ICIV and OCIV											
	2R341TRD0008	P	5R149F05046	E-7	2	A	3	BALL	MANUAL	CLOS	N/A	C
	2R341TRD0010	P	5R149F05046	E-7	2	A	3	BALL	MANUAL	CLOS	N/A	C
RH01	Residual Heat Removal Heat Exchange Control Valve (Trains A, B, and C)											
	A1RHHCV0864	A	5R169F20000	B-4	2	B	8	BUTTER	AIR	OPEN	OPEN	O
	B1RHHCV0865	A	5R169F20000	D-4	2	B	8	BUTTER	AIR	OPEN	OPEN	O
	C1RHHCV0866	A	5R169F20000	G-4	2	B	8	BUTTER	AIR	OPEN	OPEN	O
RH02	Residual Heat Removal Outlet to CVCS Letdown Valves											
	2R161XRH0066A	A	5R169F20000	A-4	2	B	4	GATE	MOTOR	OPEN	FAI	C
	2R161XRH0066B	A	5R169F20000	D-2	2	B	4	GATE	MOTOR	OPEN	FAI	C
RH03	Residual Heat Removal Pump Miniflow MOVs (Trains A, B, and C)											
	2R161XRH0067B	A	5R169F20000	D-6	2	B	4	GATE	MOTOR	CLOS	FAI	O/C
	2R161XRH0067C	A	5R169F20000	F-6	2	B	4	GATE	MOTOR	CLOS	FAI	O/C
	2R161XRH0067A	A	5R169F20000	A-6	2	B	4	GATE	MOTOR	CLOS	FAI	O/C
RH04	Residual Heat Removal Inlet Isolation MOVs (Class 1 Boundary) Trains A, B, and C											
	1R161XRH0061B	A	5R169F20000	D-8	1	A	12	GATE	MOTOR	CLOS	FAI	O/C
	1R161XRH0061C	A	5R169F20000	G-8	1	A	12	GATE	MOTOR	CLOS	FAI	O/C
	1R161XRH0060B	A	5R169F20000	D-8	1	A	12	GATE	MOTOR	CLOS	FAI	O/C
	1R161XRH0060C	A	5R169F20000	G-8	1	A	12	GATE	MOTOR	CLOS	FAI	O/C
	1R161XRH0060A	A	5R169F20000	B-8	1	A	12	GATE	MOTOR	CLOS	FAI	O/C
	1R161XRH0061A	A	5R169F20000	B-8	1	A	12	GATE	MOTOR	CLOS	FAI	O/C
RH05	Residual Heat Removal Pump Miniflow Check Valves (Trains A, B, and C)											
	2R161XRH0068A	A	5R169F20000	A-6	2	C	4	CHECK	SELF		N/A	O

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	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	2R161XRH0068B	A	5R169F20000	D-6	2	C	4	CHECK	SELF		N/A	O
	2R161XRH0068C	A	5R169F20000	F-6	2	C	4	CHECK	SELF		N/A	O
RH06	Residual Heat Removal Pump Discharge Check Valves (Trains A, B, and C)											
	2R161XRH0065A	A	5R169F20000	B-6	2	C	8	CHECK	SELF	CLOS	N/A	O
	2R161XRH0065B	A	5R169F20000	D-6	2	C	8	CHECK	SELF	CLOS	N/A	O
	2R161XRH0065C	A	5R169F20000	G-6	2	C	8	CHECK	SELF	CLOS	N/A	O
RH07	Low Head Safety Injection to RCS Hot Leg Check Valves (Trains A, B, and C)											
	1R161XRH0020A	A	5R169F20000	C-2	1	A/C	8	CHECK	SELF	CLOS	N/A	O/C
	1R161XRH0020B	A	5R169F20000	E-2	1	A/C	8	CHECK	SELF	CLOS	N/A	O/C
	1R161XRH0020C	A	5R169F20000	H-2	1	A/C	8	CHECK	SELF	CLOS	N/A	O/C
RH08	Cold Leg Injection Check Valves (Trains A, B, and C)											
	1R161XRH0032B	A	5R169F20000	D-2	1	A/C	8	CHECK	SELF	CLOS	N/A	O/C
	1R161XRH0032A	A	5R169F20000	B-2	1	A/C	8	CHECK	SELF	CLOS	N/A	O/C
	1R161XRH0032C	A	5R169F20000	G-2	1	A/C	8	CHECK	SELF	CLOS	N/A	O/C
RH09	RHR Return to RWST CIVs											
	2R161XRH0064C	P	5R169F20000	F-5	2	A	8	GATE	MANUAL	CLOS	N/A	C
	2R161XRH0064B	P	5R169F20000	D-5	2	A	8	GATE	MANUAL	CLOS	N/A	C
	2R161XRH0063B	P	5R169F20000	D-6	2	A	8	GATE	MANUAL	CLOS	N/A	C
	2R161XRH0063C	P	5R169F20000	F-6	2	A	8	GATE	MANUAL	CLOS	N/A	C
RH10	RHR Pump A, B, C Discharge Relief Valves											
	N1RHPSV3851	A	5R169F20000	C6	2	C	3	RELIEF	SELF	CLOS	N/A	O
	N1RHPSV3852	A	5R169F20000	E6	2	C	3	RELIEF	SELF	CLOS	N/A	O
	N1RHPSV3853	A	5R169F20000	H6	2	C	3	RELIEF	SELF	CLOS	N/A	O
RH11	RHR Heat Exchanger A, B, C Bypass Relief Valves											
	N1RHPSV3944	A	5R169F20000	H4	2	C	0.75	RELIEF	SELF	CLOS	N/A	O

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	N1RHPSV3934	A	5R169F20000	C4	2	C	0.75	RELIEF	SELF	CLOS	N/A	O
	N1RHPSV3943	A	5R169F20000	F4	2	C	0.75	RELIEF	SELF	CLOS	N/A	O
RM01	Reactor Make-up Water Non-essential services isolation Valves											
	C1RMFV7659	A	5R279F05033	F-7	3	B	4	GLOBE	AIR	OPEN	CLOS	C
	B1RMFV7663	A	5R279F05033	F-7	3	B	4	GLOBE	AIR	OPEN	CLOS	C
SA01	Service Air to RCB Inside Cntmt Isolation Check Valve											
	2Q101TSA0505	P	5N109F05041	D-4	2	A/C	2	CHECK	SELF	CLOS	N/A	C
SA02	Service Air to RCB Outside Cntmt Isolation Manual Valve											
	2Q101TSA0504	P	5N109F05041	C-4	2	A	2	BALL	MANUAL	CLOS	N/A	C
SB01	Steam Generator Bulk Water Sample Outside Cntmt Isolation Valves											
	A1SBFV4186	A	5S209F20	D-5	2	B	0.375	GATE	AIR	CLOS	CLOS	C
	A1SBFV4189	A	5S209F20	H-5	2	B	0.375	GATE	AIR	CLOS	CLOS	C
	B1SBFV4188	A	5S209F20	H-1	2	B	0.375	GATE	AIR	CLOS	CLOS	C
	C1SBFV4187	A	5S209F20	D-1	2	B	0.375	GATE	AIR	CLOS	CLOS	C
SB02	Steam Generator Blowdown Outside Cntmt Isolation Valves											
	A1SBFV4150	A	5S209F20001	C-5	2	B	4	GATE	AIR	CLOS	CLOS	C
	A1SBFV4153	A	5S209F20001	F-5	2	B	4	GATE	AIR	CLOS	CLOS	C
	B1SBFV4152	A	5S209F20001	F-2	2	B	4	GATE	AIR	CLOS	CLOS	C
	C1SBFV4151	A	5S209F20001	C-2	2	B	4	GATE	AIR	CLOS	CLOS	C
SD01	Starting Air Receiver Inlet Check Valves											
	3Q151XSD0004C	A	5Q159F22546	E-1	3	A/C	1	CHECK	SELF	EITH	N/A	C
	3Q151XSD0003A	A	5Q159F22546	E-7	3	A/C	1	CHECK	SELF	EITH	N/A	C
	3Q151XSD0003B	A	5Q159F22546	E-5	3	A/C	1	CHECK	SELF	EITH	N/A	C
	3Q151XSD0003C	A	5Q159F22546	E-2	3	A/C	1	CHECK	SELF	EITH	N/A	C
	3Q151XSD0004A	A	5Q159F22546	E-7	3	A/C	1	CHECK	SELF	EITH	N/A	C

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	3Q151XSD0004B	A	5Q159F22546	E-4	3	A/C	1	CHECK	SELF	EITH	N/A	C
SI01	Safety Injection System Test Line Containment Isolation Valves											
	B1SIFV3970	A	5N129F05016	F-7	2	A	0.75	GLOBE	AIR	CLOS	CLOS	C
	A1SIFV3971	A	5N129F05013	F-7	2	A	0.75	GLOBE	AIR	CLOS	CLOS	C
SI02	Accumulator Nitrogen Supply Outside Cntmt Isolation Valve (3983)											
	A1SIFV3983	A	5N129F05016	G-2	2	A	1	GLOBE	AIR	CLOS	CLOS	C
SI03	Accumulator Nitrogen Supply Inside Cntmt Isolation Check Valve (SI0058)											
	2N121TSI0058	A	5N129F05016	G-2	2	A/C	1	CHECK	SELF	CLOS	N/A	C
SI04	Reactor Water Storage Tank Clean-Up by SFPCCS Isolation Valves											
	A1SIFV3936	A	5N129F05013	F-2	2	B	3	GLOBE	AIR	EITH	CLOS	C
	B1SIFV3937	A	5N129F05013	F-2	2	B	3	GLOBE	AIR	EITH	CLOS	C
SI05	Residual Heat Exchanger Bypass Valves (Trains A, B, and C)											
	B1SIFCV0852	A	5R129F20000	E-5	2	B	8	BUTTER	AIR	CLOS	CLOS	C
	C1SIFCV0853	A	5R169F20000	H-5	2	B	8	BUTTER	AIR	CLOS	CLOS	C
	A1SIFCV0851	A	5R169F20000	C-5	2	B	8	BUTTER	AIR	CLOS	CLOS	C
SI06	Low Head Safety Injection Pump Discharge Outside Cntmt Isolation Valves (Trains A, B, and C)											
	2N121XSI0018B	A	5N129F05014	D-4	2	A	8	GATE	MOTOR	OPEN	FAI	O/C
	2N121XSI0018C	A	5N129F05015	D-4	2	A	8	GATE	MOTOR	OPEN	FAI	O/C
	2N121XSI0018A	A	5N129F05013	C-4	2	A	8	GATE	MOTOR	OPEN	FAI	O/C
SI07	Safety Injection Emergency Sump Outside Cntmt Isolation MOVs (Trains A, B, and C)											
	2N121XSI0016A	A	5N129F05013	B-4	2	A	16	GATE	MOTOR	CLOS	FAI	O/C
	2N121XSI0016B	A	5N129F05014	B-4	2	A	16	GATE	MOTOR	CLOS	FAI	O/C
	2N121XSI0016C	A	5N129F05015	B-4	2	A	16	GATE	MOTOR	CLOS	FAI	O/C
SI08	High Head Safety Injection Pump Discharge Outside Cntmt Isolation Valves (Trains A, B, and C)											
	2N121XSI0004C	A	5N129F05015	F-5	2	A	6	GATE	MOTOR	OPEN	FAI	O/C
	2N121XSI0004A	A	5N129F05013	F-5	2	A	6	GATE	MOTOR	OPEN	FAI	O/C

GROUP	GROUP DESCRIPTION						VALVE DATA			VALVE POSITIONS		
	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	2N121XSI0004B	A	5N129F05014	G-4	2	A	6	GATE	MOTOR	OPEN	FAI	O/C
SI09	High Head Safety Injection Cold Leg Isolation (Trains A, B, and C)											
	2N121XSI0006A	A	5N129F05013	E-7	2	B	6	GATE	MOTOR	OPEN	FAI	O/C
	2N121XSI0006B	A	5N129F05014	F-7	2	B	6	GATE	MOTOR	OPEN	FAI	O/C
	2N121XSI0006C	A	5N129F05015	E-7	2	B	6	GATE	MOTOR	OPEN	FAI	O/C
SI10	High Head Safety Injection Hot Leg Isolation (Trains A, B, and C)											
	2N121XSI0008C	A	5N129F05015	F-7	2	B	6	GATE	MOTOR	CLOS	FAI	O/C
	2N121XSI0008A	A	5N129F05013	F-7	2	B	6	GATE	MOTOR	CLOS	FAI	O/C
	2N121XSI0008B	A	5N129F05014	G-7	2	B	6	GATE	MOTOR	CLOS	FAI	O/C
SI11	Residual Heat Removal Heat Exchanger Return to Hot Leg MOV (Trains A, B, and C)											
	2R161XRH0019B	A	5R169F20000	E-3	2	B	8	GATE	MOTOR	CLOS	FAI	O
	2R161XRH0019C	A	5R169F20000	H-3	2	B	8	GATE	MOTOR	CLOS	FAI	O
	2R161XRH0019A	A	5R169F20000	C-3	2	B	8	GATE	MOTOR	CLOS	FAI	O
SI12	Cold Leg Injection MOVs (Trains A, B, C)											
	2R161XRH0031A	A	5R169F20000	B-3	2	B	8	GATE	MOTOR	OPEN	FAI	O/C
	2R161XRH0031B	A	5R169F20000	D-3	2	B	8	GATE	MOTOR	OPEN	FAI	O/C
	2R161XRH0031C	A	5R169F20000	G-3	2	B	8	GATE	MOTOR	OPEN	FAI	O/C
SI13	High Head Safety Injection Pump Recirc Isolation											
	2N121TSI0012B	A	5N129F05014	H-3	2	B	2	DIAPHR	MOTOR	OPEN	FAI	O/C
	2N121TSI0012C	A	5N129F05015	G-3	2	B	2	DIAPHR	MOTOR	OPEN	FAI	O/C
	2N121TSI0012A	A	5N129F05013	F-4	2	B	2	DIAPHR	MOTOR	OPEN	FAI	O/C
	2N121TSI0011C	A	5N129F05015	G-4	2	B	2	DIAPHR	MOTOR	OPEN	FAI	O/C
	2N121TSI0011B	A	5N129F05014	H-3	2	B	2	DIAPHR	MOTOR	OPEN	FAI	O/C
	2N121TSI0011A	A	5N129F05013	F-4	2	B	2	DIAPHR	MOTOR	OPEN	FAI	O/C
SI14	Low Head Safety Injection Pump Recirc Isolation											

GROUP	GROUP DESCRIPTION						VALVE DATA			VALVE POSITIONS		
	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	2N121TSI0013B	A	5N129F05014	E-3	2	B	2	DIAPHR	MOTOR	OPEN	FAI	O/C
	2N121TSI0013C	A	5N129F05015	D-3	2	B	2	DIAPHR	MOTOR	OPEN	FAI	O/C
	2N121TSI0014A	A	5N129F05013	D-3	2	B	2	DIAPHR	MOTOR	OPEN	FAI	O/C
	2N121TSI0014B	A	5N129F05014	E-3	2	B	2	DIAPHR	MOTOR	OPEN	FAI	O/C
	2N121TSI0014C	A	5N129F05015	D-3	2	B	2	DIAPHR	MOTOR	OPEN	FAI	O/C
	2N121TSI0013A	A	5N129F05013	D-3	2	B	2	DIAPHR	MOTOR	OPEN	FAI	O/C
SI15	Safety Injection System Reactor Water Storage Tank Isolation											
	2N121XSI0001A	A	5N129F05013	G-3	2	B	16	GATE	MOTOR	OPEN	FAI	O/C
	2N121XSI0001B	A	5N129F05014	H-2	2	B	16	GATE	MOTOR	OPEN	FAI	O/C
	2N121XSI0001C	A	5N129F05015	H-2	2	B	16	GATE	MOTOR	OPEN	FAI	O/C
SI16	Accumulator Nitrogen Vent Valves (Trains A, B, and C)											
	B1SIPV3930	A	5N129F05016	G-4	2	B	1	GLOBE	SOLENO	CLOS	CLOS	O
	A1SIPV3928	A	5N129F05016	C-4	2	B	1	GLOBE	SOLENO	CLOS	CLOS	O
	C1SIPV3929	A	5N129F05016	E-4	2	B	1	GLOBE	SOLENO	CLOS	CLOS	O
SI17	Accumulator Nitrogen Vent Back-Up Valve (899)											
	B1SIHV0899	A	5N129F05016	F-2	2	B	1	GLOBE	SOLENO	CLOS	CLOS	O
SI18	High Head Safety Injection Pump Discharge Inside Cntmt Isolation Valves (Trains A, B, and C)											
	2N121XSI0005C	A	5N129F05015	F-5	2	A/C	6	CHECK	SELF	CLOS	N/A	O/C
	2N121XSI0005B	A	5N129F05014	G-4	2	A/C	6	CHECK	SELF	CLOS	N/A	O/C
	2N121XSI0005A	A	5N129F05013	F-6	2	A/C	6	CHECK	SELF	CLOS	N/A	O/C
SI19	High Head Safety Injection Pump Discharge Check to Cold Leg (Class 1 Boundary) (Trains A, B, and C)											
	1N121XSI0007A	A	5N129F05013		1	A/C	6	CHECK	SELF	CLOS	N/A	O/C
	1N121XSI0007B	A	5N129F05014	F-7	1	A/C	6	CHECK	SELF	CLOS	N/A	O/C
	1N121XSI0007C	A	5N129F05015	E-7	1	A/C	6	CHECK	SELF	CLOS	N/A	O/C
SI20	High Head Safety Injection Pump Discharge Check to Hot Leg (Class 1 Boundary) (Trains A, B, and C)											

GROUP	GROUP DESCRIPTION						VALVE DATA			VALVE POSITIONS		
	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	1N121XSI0009C	A	5N129F05015	F-7	1	A/C	6	CHECK	SELF	CLOS	N/A	O/C
	1N121XSI0009A	A	5N129F05013	F-7	1	A/C	6	CHECK	SELF	CLOS	N/A	O/C
	1N121XSI0009B	A	5N129F05014	G-7	1	A/C	6	CHECK	SELF	CLOS	N/A	O/C
SI21	Low Head Safety Injection Pump Discharge Inside Cntmt Isolation Valves (Trains A, B, and C)											
	2N121XSI0030C	A	5N129F05015	D-4	2	A/C	8	CHECK	SELF	CLOS	N/A	O/C
	2N121XSI0030A	A	5N129F05013	D-5	2	A/C	8	CHECK	SELF	CLOS	N/A	O/C
	2N121XSI0030B	A	5N129F05014	D-4	2	A/C	8	CHECK	SELF	CLOS	N/A	O/C
SI22	Safety Injection System Pumps Discharge Check to Hot Leg (Class 1 Boundary) (Trains A, B, and C)											
	1N121XSI0010A	A	5N129F05013	F-8	1	A/C	8	CHECK	SELF	CLOS	N/A	O/C
	1N121XSI0010B	A	5N129F05014	G-8	1	A/C	8	CHECK	SELF	CLOS	N/A	O/C
	1N121XSI0010C	A	5N129F05015	F-8	1	A/C	8	CHECK	SELF	CLOS	N/A	O/C
SI23	Accumulator to Cold Leg Inboard Check Valves (Trains A, B, and C)											
	1N121XSI0038C	A	5N129F05016	B-7	1	A/C	12	CHECK	SELF	CLOS	N/A	O/C
	1N121XSI0038B	A	5N129F05016	D-7	1	A/C	12	CHECK	SELF	CLOS	N/A	O/C
	1N121XSI0038A	A	5N129F05016	F-7	1	A/C	12	CHECK	SELF	CLOS	N/A	O/C
SI24	Accumulator Tank Discharge MOVs (Trains A, B, and C)											
	2N121XSI0039A	A	5N129F05016	F-5	2	B	12	GATE	MOTOR	OPEN	FAI	O/C
	2N121XSI0039B	A	5N129F05016	D-5	2	B	12	GATE	MOTOR	OPEN	FAI	O/C
	2N121XSI0039C	A	5N129F05016	B-5	2	B	12	GATE	MOTOR	OPEN	FAI	O/C
SI25	Safety Injection Pumps Suction Check Valves (Trains A, B, and C)											
	2N121XSI0002A	A	5N129F05013	G-3	2	C	16	CHECK	SELF	CLOS	N/A	O/C
	2N121XSI0002C	A	5N129F05015	H-2	2	C	16	CHECK	SELF	CLOS	N/A	O/C
	2N121XSI0002B	A	5N129F05014	H-2	2	C	16	CHECK	SELF	CLOS	N/A	O/C
SI26	Accumulator Nitrogen Vent Header Bleed Valve (HCV-0900)											
	A1SIHCV0900	A	5N129F05016	G-2	2	B	1	GLOBE	SOLENO	CLOS	CLOS	O

GROUP	GROUP DESCRIPTION						VALVE DATA			VALVE POSITIONS		
	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
SI27	Accumulator to Cold Leg Outboard Check Valves (Trains A, B, and C)											
	1N121XSI0046B	A	5N129F05016	D-7	1	A/C	12	CHECK	SELF	CLOS	N/A	O/C
	1N121XSI0046C	A	5N129F05016	B-7	1	A/C	12	CHECK	SELF	CLOS	N/A	O/C
	1N121XSI0046A	A	5N129F05016	F-6	1	A/C	12	CHECK	SELF	CLOS	N/A	O/C
SI28	Safety Injection Train A, B, C Pumps Suction Header Relief Valves											
	N1SIPSV3935	A	5N129F05013	C2	2	C	0.75	RELIEF	SELF	CLOS	N/A	O
	N1SIPSV3939	A	5N129F05014	D2	2	C	0.75	RELIEF	SELF	CLOS	N/A	O
	N1SIPSV3941	A	5N129F05015	C2	2	C	0.75	RELIEF	SELF	CLOS	N/A	O
SI29	HHSI Pump A, B, C Disch to Loop A, B, C Hot/Cold Leg Relief Valves											
	N1SIPSV3942	A	5N129F05015	F6	2	C	0.75	RELIEF	SELF	CLOS	N/A	O
	N1SIPSV3940	A	5N129F05014	F6	2	C	0.75	RELIEF	SELF	CLOS	N/A	O
	N1SIPSV3938	A	5N129F05013	G6	2	C	0.75	RELIEF	SELF	CLOS	N/A	O
SI30	Safety Injection Accumulator A, B, C Relief Valves											
	N1SIPSV3977	A	5N129F05016	C4	2	C	1	RELIEF	SELF	CLOS	N/A	O
	N1SIPSV3981	A	5N129F05016	G4	2	C	1	RELIEF	SELF	CLOS	N/A	O
	N1SIPSV3980	A	5N129F05016	E4	2	C	1	RELIEF	SELF	CLOS	N/A	O
SL1	High Pressure Sludge Lancing CIVs											
	2S201TSL0002	P	5S129F05057	B-5	2	A	2	GATE	MANUAL	CLOS	N/A	C
	2S201TSL0004	P	5S129F05057	B-6	2	A	2	GATE	MANUAL	CLOS	N/A	C
SL2	Low Pressure Sludge Lancing CIVs											
	2S201TSL0029	P	5S129F05057	F-5	2	A	6	GATE	MANUAL	CLOS	N/A	C
	2S201TSL0027	P	5S129F05057	F-6	2	A	6	GATE	MANUAL	CLOS	N/A	C
	2S201TSL0014	P	5S129F05057	D-6	2	A	6	GATE	MANUAL	CLOS	N/A	C
	2S201TSL0012	P	5S129F05057	D-5	2	A	6	GATE	MANUAL	CLOS	N/A	C
WL01	RCDT Vent Outside Containment Isolation Valve											

GROUP	GROUP DESCRIPTION						VALVE DATA			VALVE POSITIONS		
	TAGTPNS	Act/Pass	PID #	Coord.	QClass	IST Cat	Size	Type	Actuator	Normal	Failsafe	Safety Func.
	B1WLFV4919	A	5R309F05022	G-5	2	A	1	GLOBE	AIR	OPEN	CLOS	C
WL02	RCDT To LWPS Outside Containment Isolation Valve											
	B1WLFV4913	A	5R309F05022	F-3	2	A	3	GLOBE	AIR	OPEN	CLOS	C
WL03	RCDT To LWPS Inside Containment Isolation Valve											
	2R301TWL0312	A	5R309F05022	E-3	2	A	3	GATE	MOTOR	OPEN	FAI	C
WL04	RCDT Vent Inside Containment Isolation Valve											
	A1WLFV4920	A	5R309F05022	G-6	2	A	1	GLOBE	SOLENO	OPEN	CLOS	C
XC01	Reactor Containment Personal Air-lock Safety Check Valves (XC-48,49)											
	2C261XXC0049	A	5C269F05060	C-7	2	A/C	1	CHECK	SELF	CLOS	N/A	
	2C261XXC0048	A	5C269F05060	C-7	2	A/C	1	CHECK	SELF	CLOS	N/A	
XC02	Reactor Containment Air-lock Air Supply Containment Isolation Valves (FV1025, 26,27,28)											
	A1XCFV1027	A	5C269F05060	C-4	2	A	0.5	GLOBE	SOLENO	OPEN	CLOS	C
	A1XCFV1028	A	5C269F05060	C-4	2	A	0.5	GLOBE	SOLENO	OPEN	CLOS	C
	A1XCFV1026	A	5C269F05060	F-4	2	A	0.5	GLOBE	SOLENO	OPEN	CLOS	C
	A1XCFV1025	A	5C269F05060	G-4	2	A	0.5	GLOBE	SOLENO	OPEN	CLOS	C

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
AF01	Auxiliary Feedwater Supply to Steam Generator Inside Cntmt Isolation Check Valves				
	CV-O	High	CS	CS (CSJ-02)	Check Valve Open Exercise
AF02	Auxiliary Feedwater Supply to Steam Generator Outside Cntmt Isolation Stop Check MOVs				
	CV-C	High	Q	Q	Check Valve Close Exercise
	CV-O	High	CS	CS (CSJ-01)	Check Valve Open Exercise
	PI	High	2Y	2Y	Position Indication
	ST-C	High	Q	Q	Stroke Time Measurement - Close
	ST-O	High	Q	Q	Stroke Time Measurement - Open
AF03	Auxiliary Feedwater Supply to Steam Generator Flow Regulating MOVs				
	PI	High	2Y	2Y	Position Indication
	ST-C	High	Q	Q	Stroke Time Measurement - Close
	ST-O	High	Q	Q	Stroke Time Measurement - Open
AF04	Auxiliary Feedwater Turbine Trip and Trottle Valve (MS0514)				
	PI	High	2Y	2Y	Position Indication
	ST-C	High	Q	Q	Stroke Time Measurement - Close
	ST-O	High	Q	Q	Stroke Time Measurement - Open
AF05	Main Steam to Auxiliary Feedwater Turbine Warm-up Valve				
	FS-C	Low	Q	18MO	Fail Safe Test - Close
	PI	Low	2Y	18MO	Position Indication
	ST-C	Low	Q	18MO	Stroke Time Measurement - Close
	ST-O	Low	Q	18MO	Stroke Time Measurement - Open
AF06	Auxiliary Feedwater Pump Discharge Cross-Tie Valves				
	FS-C	Low	Q	54MO	Fail Safe Test - Close
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
AF07	Auxiliary Feedwater Auto Recirc Valves				
	CV-O	High	Q	Q	Check Valve Open Exercise
	CV-OP	High	Q	Q	Check Valve Partial Open Exercise
AF08	Main Steam to AF Turbine Suction Stop Check MOV (MS0143)				
	CV-O	Medium	Q	R	Check Valve Open Exercise
	PI	Medium	2Y	2Y	Position Indication
	ST-C	Medium	Q	R	Stroke Time Measurement - Close
	ST-O	Medium	Q	R	Stroke Time Measurement - Open
AF09	Auxiliary Feedwater Pump Discharge Cross-Tie Valve (D train)				
	FS-C	Low	Q	R	Fail Safe Test - Close
	PI	Low	2Y	2Y	Position Indication

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
	ST-C	Low	Q	R	Stroke Time Measurement - Close
	ST-O	Low	Q	R	Stroke Time Measurement - Open
AP01	RCS Hot Leg Sample to PASS Lab OCIVs				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y (VRR-01)	Position Indication
	ST-C	Low	Q	R (VRR-02)	Stroke Time Measurement - Close
AP02	Cntmt Normal Sump to PASS Lab OCIVs				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
AP03	RHR Sample to PASS Lab OCIVs				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
AP04	PASS Waste Collection Unit Return to Pressurizer Relief Tank OCIV				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
AP05	Containment Air Sample Supply and Return to PASS Lab OCIVs				
	FS-C	Low	Q	3YR	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
BA01	Breathing Air System Inside Cntmt Isolation Check Valve				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
BA02	Breathing Air System Outside Cntmt Isolation Manual Valve				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
CC01	Thermal Relief for Penetration M-40 CCW return for the RCPs				
	CV-C	Low	RF	R	Check Valve Close Exercise
	CV-O	Low	RF	R	Check Valve Open Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
CC02	CCW Supply to the RCP Thermal Barriers (Double inlet check valves)				
	DA	Low	RF	6YR	Disassemble and Inspect

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
CC03	Penetration M-40 CCW return for the RCPs				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
CC04	RHR Heat Exchanger - CCW Outlet Valves				
	FS-O	High	Q	Q	Fail Safe Test - Open
	PI	High	2Y	2Y	Position Indication
	ST-O	High	Q	Q	Stroke Time Measurement - Open
CC05	Common Suction Header Isolation Valves (Trains A, B, & C) MOVs				
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CC06	Common Supply Header Isolation Valves (Trains A, B, & C)				
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CC07	CCW Heat Exchanger Outlet MOVs (Trains A, B, and C)				
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CC08	CCW Heat Exchanger Bypass MOVs (Trains A, B, and C)				
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CC09	CCW return from the RCFCs, Inside Containment Isolation Valves (Trains A, B, and C)				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CC09A	CCW return from the RCFCs, Outside Containment Isolation Valves (Trains A, B, and C)				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CC10	CCW Supply (OCIV) to RHR Pump and Heat Exchanger - Trains A, B, and C				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CC11	CCW Supply (OCIV) to Reactor Containment Fan Coolers - Trains A, B, and C				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CC12	CCW Return from RHR Pump and Heat Exchanger - Trains A, B, and C				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CC13	Chilled Water Return from RCFCs Outside Cntmt. Isolation MOV (Trains A, B, and C)				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CC14	Chilled Water Supply to RCFCs Outside Cntmt. Isolation MOV (Trains A, B, and C)				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CC15	CCW Supply Header to Spent Fuel Pool Heat Exchanger, First and Second Isolation				
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
CC16	CCW Supply Header to Non-Safety Loads, First and Second Isolation				
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	CS	3YR	Stroke Time Measurement - Close
CC17	CCW Supply to Excess Letdown Heat Exchanger Isolation MOV				
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
CC18	CCW Supply Header Isolation to Charging Pumps (Trains A, B, and C)				
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CC19	CCW Return Isolation from Charging Pumps (Trains A, B, and C)				

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CC20	CCW Supply to RCDT Ht. Exch. and Excess Letdown				
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
CC21	CCW Supply to RCDT Ht. Exch.				
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
CC22	CCW Supply to RCP Coolers Outside Cntmt Isolation MOVs				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
CC23	CCW Return from RCP Coolers, Cntmt Isolation MOVs				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
CC24	Chilled Water Return for the RCFCs, Outside Cntmt Isolation Valve (Trains A, B, and C)				
	FS-C	Low	Q	54MO	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
CC25	CCW Supply Header to Post Accident Sampling System, First and Second Isolation				
	FS-C	Low	Q	3YR	Fail Safe Test - Close
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
CC26	CCW Common Return Header to CCW Pump Suction Check Valve (Trains A, B, and C)				
	CV-C	Low	Q	54MO	Check Valve Close Exercise
	CV-O	Low	Q	54MO	Check Valve Open Exercise
CC27	CCW Pump Discharge Check Valve to Common Supply Header (Trains A, B, and C)				
	CV-O	Low	Q	54MO	Check Valve Open Exercise
CC28	CCW Supply to RCFCs Inside Cntmt Isolation Check Valve (Trains A, B, and C)				
	CV-C	Low	APP J	APP J	Check Valve Close Exercise
	CV-O	Low	Q	54MO	Check Valve Open Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
CC29	CCW Supply to RHR Pump and Heat Exchanger Inside Cntmt Isolation Check Valve (Trains A, B, and C)				

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
	CV-C	High	APP J	APP J (VRR-03)	Check Valve Close Exercise
	CV-O	High	Q	Q	Check Valve Open Exercise
	LR-CIV-AL	High	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
CC30	CCW Return for RCDT Heat Exchanger Check Valves				
	DA	Low	RF	3YR	Disassemble and Inspect
CC31	CCW Return for Excess Letdown Heat Exchanger Check Valves				
	DA	Low	RF	3YR	Disassemble and Inspect
CC32	CCW Supply to RCPs Inside Containment Isolation Check Valve				
	CV-C	Low	RF	R	Check Valve Close Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
CC33	RCP Thermal Barrier Leak Isolation Valves				
	FSE	Low	RF	6YR	Full Stroke Exercise (Manual Valves)
	SP	Low	RF	6YR	Setpoint Verification
CC34	Cross Connect Valves for CCW Supply and Return for Charging Pumps				
	FS-C	Low	Q	3YR	Fail Safe Test - Close
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
CH01	EAB Control Room Envelope Air Handling Unit Outlet Temperature Valve (Trains A, B, and C)				
	FS-O	Low	Q	54MO	Fail Safe Test - Open
	PI	Low	2Y	54MO	Position Indication
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CH02	EAB Main Supply Air Handling Unit Outlet Temperature Valve (Trains A, B, and C)				
	FS-O	Low	Q	54MO	Fail Safe Test - Open
	PI	Low	2Y	54MO	Position Indication
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CM01	RCB Air Sample Select Valves for Cntmt Hydrogen Monitoring System				
	PI	Low	2Y	6YR	Position Indication
	ST-C	Low	Q	6YR	Stroke Time Measurement - Close
	ST-O	Low	Q	6YR	Stroke Time Measurement - Open
CM02	Cntmt Hydrogen Monitoring System Inside and Outside CIVs				
	FS-C	Low	Q	6YR	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	6YR	Position Indication
	ST-C	Low	Q	6YR	Stroke Time Measurement - Close
	ST-O	Low	Q	6YR	Stroke Time Measurement - Open
CS01	Containment Spray Pump Discharge Outside Cntmt Isolation MOVs				

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
CS02	Containment Spray Header Inside Cntmt Isolation Check Valves				
	DA	Low	RF	6YR	Disassemble and Inspect
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
CV01	Reactor Coolant Auxiliary Spray Valve				
	PI	Medium	2Y	2Y	Position Indication
	ST-C	Medium	CS	R	Stroke Time Measurement - Close
	ST-O	Medium	CS	R	Stroke Time Measurement - Open
CV02	Centrifugal Charging Pump Minimum Recirc. Control Valves				
	FS-O	Low	Q	3YR	Fail Safe Test - Open
	PI	Low	2Y	3YR	Position Indication
	ST-O	Low	Q	3YR	Stroke Time Measurement - Open
CV03	RCS Letdown Line Inside Cntmt Isolation Bypass Check Valve (CV0022)				
	CV-C	Low	RF	R	Check Valve Close Exercise
	CV-O	Low	RF	R	Check Valve Open Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
CV04	RCS Seal Water Return Inside Cntmt Isolation Bypass Check Valve (CV0078)				
	CV-C	Low	RF	R	Check Valve Close Exercise
	CV-O	Low	RF	R	Check Valve Open Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
CV05	(CV0346,351) BAT Pump recirc valves				
	CV-O	Low	Q	3YR	Check Valve Open Exercise
CV06	RCP Seal Injection Check Valve (Class 1 Boundary Isolation)				
	CV-C	Low	R	6YR	Check Valve Close Exercise
	CV-O	Low	Q	6YR	Check Valve Open Exercise
CV07	Seal Injection to RCPs Inside Cntmt Isolation Check Valves				
	CV-C	Low	RF	6YR	Check Valve Close Exercise
	CV-O	Low	Q	6YR	Check Valve Open Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
CV08	Boric Acid Polishing Return to Boric Acid Tank				
	CV-C	Low	Q	3YR	Check Valve Close Exercise
CV09	Centrifugal Charging Pump Minimum Recirc. Check Valves				
	CV-O	Low	Q	3YR	Check Valve Open Exercise

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
CV10	Reactor Coolant Auxiliary Spray Inlet Check Valve (CV0009)				
	CV-O	Medium	CS	R	Check Valve Open Exercise
CV11	CVCS SEAL WATER INJECTION FLOW CONTROL VALVE				
	FS-O	Low	CS	R	Fail Safe Test - Open
	PI	Low	2Y	2Y	Position Indication
	ST-O	Low	CS	R	Stroke Time Measurement - Open
CV12	Letdown Orifice Header Isolation Valve				
	FS-C	Low	CS	R	Fail Safe Test - Close
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	CS	R	Stroke Time Measurement - Close
CV13	RCS Charging Flow Control Valve				
	FS-O	Medium	CS	R	Fail Safe Test - Open
	PI	Medium	2Y	2Y	Position Indication
	ST-O	Medium	CS	R	Stroke Time Measurement - Open
CV14	Manual Alternate Borate Check Valve				
	CV-C	Low	RF	R	Check Valve Close Exercise
	CV-O	Low	RF	R	Check Valve Open Exercise
CV15	Charging Header Check Valve (CV671)				
	CV-C	Low	RF	R	Check Valve Close Exercise
	CV-O	Low	Q	R	Check Valve Open Exercise
CV16	Boric Acid Supply to Concentrated BA Polishing Demineralizer Isolation Valves				
	FS-C	Low	Q	3YR	Fail Safe Test - Close
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
CV19	RCS Charging Outside Cntmt Isolation MOV				
	LR-CIV-AL	Medium	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Medium	2Y	2Y	Position Indication
	ST-C	Medium	CS	R	Stroke Time Measurement - Close
	ST-O	Medium	CS	R	Stroke Time Measurement - Open
CV20	RCS Letdown Isolation (Class 1 Boundary Isolation)				
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	CS	3YR	Stroke Time Measurement - Close
CV21	Centrifugal Charging Pump Discharge Isolation MOVs				
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
	ST-O	Low	Q	3YR	Stroke Time Measurement - Open

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
CV22	Volume Control Tank Outlet Isolation MOVs				
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	CS	3YR	Stroke Time Measurement - Close
CV23	Reactor Water Storage Tank to Charging Pump Suction Header Isolation MOVs				
	PI	Low	2Y	3YR	Position Indication
	ST-O	Low	CS	3YR	Stroke Time Measurement - Open
CV24	Alternate Boric Acid Make-Up Supply Isolation MOV (CV0218)				
	PI	Low	2Y	2Y	Position Indication
	ST-O	Low	Q	R	Stroke Time Measurement - Open
CV25	RCS Normal and Alternate Charging Flow Isolation MOVs				
	PI	Medium	2Y	3YR	Position Indication
	ST-C	Medium	CS	3YR	Stroke Time Measurement - Close
	ST-O	Medium	CS	3YR	Stroke Time Measurement - Open
CV26	RCS Letdown Inside and Outside Cntmt Isolation MOVs				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	CS	3YR	Stroke Time Measurement - Close
CV27	RCP Seal Injection Outside Cntmt Isolation MOVs				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	6YR	Position Indication
	ST-C	Low	CS	6YR	Stroke Time Measurement - Close
CV29	RCP Seal Water Return Inside and Outside Cntmt Isolation MOVs				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	CS	3YR	Stroke Time Measurement - Close
CV30	RCS Excess Letdown Heat Exchanger Inlet Isolation MOVs (Class 1 Boundary Isolation)				
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
CV32	Charging Pump B Discharge Bypass Control Valve				
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
	ST-O	Low	Q	R	Stroke Time Measurement - Open
CV33	Centrifugal Charging Pump Discharge Check Valves				
	CV-C	Low	Q	3YR	Check Valve Close Exercise
	CV-O	Low	Q	3YR	Check Valve Open Exercise
CV34	(CV0334) check valve				

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
	CV-O	Low	CS	R	Check Valve Open Exercise
CV35	RC Filters out to RHR Outside Cntmt Isolation Manual Valve				
	LR-CIV-AL	APP J	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
CV37	Charging Header Check Valve				
	CV-C	Low	RF	R	Check Valve Close Exercise
	CV-O	Low	Q	R	Check Valve Open Exercise
CV38	RCS Normal and Alternate Charging Check Valves (Class 1 Boundary Valves)				
	CV-C	Low	RF	3YR	Check Valve Close Exercise
	CV-O	Low	CS	3YR	Check Valve Open Exercise
CV40	RCS Charging Inside Cntmt Isolation Check Valve.				
	CV-C	Low	RF	R	Check Valve Close Exercise
	CV-O	Low	Q	R	Check Valve Open Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
CV41	Alternate Boric Acid Make-Up Supply Isolation Check Valve (CV0217)				
	CV-O	Low	CS	R	Check Valve Open Exercise
CV42	Boric Acid Pump Discharge Check Valves (CV349,338)				
	CV-C	Low	Q	3YR	Check Valve Close Exercise
	CV-O	Low	Q	3YR	Check Valve Open Exercise
CV43	RC Filters out to RHR Inside Cntmt Isolation Check Valve				
	LR-CIV-AL	APP J	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
CV44	Reactor Water Storage Tank to Charging Pump Suction Header Isolation Check Valve				
	CV-O	Low	CS	R	Check Valve Open Exercise
DW01	Demineralizer Water to the RCB Inside Cntmt Isolation Check Valve				
	LR-CIV-AL	APP J	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
DW02	Demineralizer Water to the RCB Outside Cntmt Isolation Manual Valve				
	LR-CIV-AL	APP J	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
ED01	Containment Normal Sump Discharge Outside Cntmt Isolation Valve (FV7800)				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
ED02	Containment Normal Sump Discharge Inside Cntmt Isolation MOV (ED0064)				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
EW01	Essential Cooling Water Blowdown Isolation Valve (Trains A, B, and C)				
	FS-C	Low	Q	54MO	Fail Safe Test - Close
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
EW02	Essential Cooling Water Pump Discharge Vent Check Valve (Trains A, B, and C)				
	DA	Low	RF	54MO	Disassemble and Inspect
EW03	ECW Screen Wash Booster Pump Discharge Check Valve (Trains A, B, and C)				
	CV-O	Low	Q	54MO	Check Valve Open Exercise
EW04	Essential Cooling Water Pump Discharge Strainer Emergency Backflush Check Valve (Trains A, B, and C)				
	CV-O	Low	Q	54MO	Check Valve Open Exercise
	DA	Low	RF	54MO	Disassemble and Inspect
EW05	Essential Cooling Water Pump Discharge MOV (Trains A, B, and C)				
	PI	Medium	2Y	54MO	Position Indication
	ST-O	Medium	Q	54MO	Stroke Time Measurement - Open
EW06	ECW Self-Cleaning Strainer Backflush Throttle Valve (Manual)				
	FSE	Low	Q	54MO	Full Stroke Exercise (Manual Valves)
EW07	ECW Self-Cleaning Strainer Emergency Backflush Manual Valve				
	FSE	Low	Q	54MO	Full Stroke Exercise (Manual Valves)
EW08	Essential Cooling Water Pump Discharge Check Valve (Trains A, B, and C)				
	CV-O	High	Q	Q	Check Valve Open Exercise
EW09	ECW Screen Wash Pump Discharge Valve (Trains A, B, and C)				
	FS-O	Low	Q	54MO	Fail Safe Test - Open
	PI	Low	2Y	54MO	Position Indication
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
FC01	SFP Pump Discharge Reactor Cavity ICIV (Manual)				
	LR-CIV-AL	APP J	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
FC02	SFP Pump Cooling Supply and Return from In-Cntmt Storage Area CIV (Manual)				
	LR-CIV-AL	APP J	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
FP01	Fire Protection to the RCB Inside Cntmt Isolation Check Valve				
	CV-C	Low	RF	APP J	Check Valve Close Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
FP02	Fire Protection to the RCB Outside Cntmt Isolation MOV				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
FW01	Feedwater to the Steam Generator Isolation Valves				
	FS-C	Low	CS	6YR	Fail Safe Test - Close
	PI	Low	2Y	6YR	Position Indication
	PSE	Low	Q	6YR	Partial Stroke Exercise
	ST-C-A	Low	CS	6YR	Stroke Time Measurement - Close (A Train)
	ST-C-B	Low	CS	6YR	Stroke Time Measurement - Close (B Train)
FW02	Feedwater flow control valves				
	FS-C	Low	CS	6YR	Fail Safe Test - Close
	PI	Low	2Y	6YR	Position Indication
	ST-C-A	Low	CS	6YR	Stroke Time Measurement - Close (A Train)
	ST-C-B	Low	CS	6YR	Stroke Time Measurement - Close (B Train)
FW03	Feedwater Bypass Flow Control Valves				
	FS-C	Low	CS	6YR	Fail Safe Test - Close
	PI	Low	2Y	6YR	Position Indication
	ST-C-A	Low	CS	6YR	Stroke Time Measurement - Close (A Train)
	ST-C-B	Low	CS	6YR	Stroke Time Measurement - Close (B Train)
FW04	Steam Generator Feedwater Inlet Isolation Bypass Valves				
	FS-C	Low	Q	6YR	Fail Safe Test - Close
	PI	Low	2Y	6YR	Position Indication
	ST-C-A	Low	Q	6YR	Stroke Time Measurement - Close (A Train)
	ST-C-B	Low	Q	6YR	Stroke Time Measurement - Close (B Train)
FW05	Steam Generator Preheater Bypass Valves				
	FS-C	Low	CS	6YR	Fail Safe Test - Close
	PI	Low	2Y	6YR	Position Indication
	ST-C-A	Low	CS	6YR	Stroke Time Measurement - Close (A Train)
	ST-C-B	Low	CS	6YR	Stroke Time Measurement - Close (B Train)
HC01	RCB Supplemental Purge Supply and Return Inside Cntmt Isolation MOVs				
	LR-CIV-AL	Medium	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Medium	2Y	3YR	Position Indication
	ST-C	Medium	Q	3YR	Stroke Time Measurement - Close
HC02	RCB Supplemental Purge Supply and Return Outside Cntmt Isolation AOVs				
	FS-C	Medium	Q	3YR	Fail Safe Test - Close
	LR-CIV-AL	Medium	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Medium	2Y	3YR	Position Indication
	ST-C	Medium	Q	3YR	Stroke Time Measurement - Close
HC03	RCB Normal Purge Supply and Exhaust Cntmt Isolation (48") MOVs				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
	PI	Low	2Y	6YR	Position Indication
	ST-C	Low	CS	6YR	Stroke Time Measurement - Close
IA01	Instrument Air to RCB Inside Cntmt Isolation Check Valve (IA0541)				
	CV-C	Low	RF	APP J	Check Valve Close Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
IA02	Instrument Air to RCB Outside Cntmt Isolation Valve (IA8565)				
	FS-C	Low	CS	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	CS	R	Stroke Time Measurement - Close
MS01	Main Steam Isolation Valves				
	FS-C	Low	CS	6YR	Fail Safe Test - Close
	PI	Low	2Y	6YR	Position Indication
	ST-C-A	Low	CS	6YR	Stroke Time Measurement - Close (A Train)
	ST-C-B	Low	CS	6YR	Stroke Time Measurement - Close (B Train)
MS02	Main Steam Safety Valves				
	SP	Medium	RF	5YR	Setpoint Verification
	SP	Medium	RF	5YR	Setpoint Verification
	SP	Medium	RF	5YR	Setpoint Verification
	SP	Medium	RF	5YR	Setpoint Verification
	SP	Medium	RF	5YR	Setpoint Verification
	SP	Medium	RF	5YR	Setpoint Verification
	SP	Medium	RF	5YR	Setpoint Verification
	SP	Medium	RF	5YR	Setpoint Verification
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	SP	Medium	RF	5YR	Setpoint Verification
	SP	Medium	RF	5YR	Setpoint Verification
	SP	Medium	RF	5YR	Setpoint Verification
	SP	Medium	RF	5YR	Setpoint Verification
	SP	Medium	RF	5YR	Setpoint Verification
MS03	Main Steam Power Operated Relief Valves				
	FS-C	High	Q	Q	Fail Safe Test - Close

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
	PI	High	2Y	2Y	Position Indication
	ST-C	High	Q	Q	Stroke Time Measurement - Close
	ST-O	High	Q	Q	Stroke Time Measurement - Open
MS04	Main Steam Bypass Isolation Valves				
	FS-C	Low	Q	6YR	Fail Safe Test - Close
	PI	Low	2Y	6YR	Position Indication
	ST-C-A	Low	Q	6YR	Stroke Time Measurement - Close (A Train)
	ST-C-B	Low	Q	6YR	Stroke Time Measurement - Close (B Train)
PO01	RCP Motor Oil Return system				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
PS01	Pressurizer Vapor Space Sample Inside Cntmt Isolation Valve (4450)				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
PS02	RCS Pressurizer and Hot Leg Sample ICIVs				
	FS-C	Low	Q	54MO	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
PS03	RHR and Accumulator Sample ICIVs				
	FS-C	Low	Q	3YR	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
PS04	Pressurizer Liquid Sample OCIV				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
PS05	Pressurizer Vapor Space Sample OCIV				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
PS07	Primary sampling OCIVs (FV4461 and FV4466, FV 4456)				
	FS-C	Low	Q	3YR	Fail Safe Test - Close

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
RA01	RCB Atmosphere Rad Monitor Inside and Outside Cntmt Isolation Valves				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	6YR	Position Indication
	ST-C	Low	Q	6YR	Stroke Time Measurement - Close
RC01	Pressurizer Relief Tank Vent to Gaseous Waste Processing System Outside Cntmt Isolation Valve (3652)				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
RC02	Reactor Make-up Water to RCP Standpipe and PRT OCIV (3651)				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
RC03	RCS Pressurizer Safety Valves				
	SP	Medium	RF	R	Setpoint Verification
	SP	Medium	RF	R	Setpoint Verification
	SP	Medium	RF	R	Setpoint Verification
RC04	RCS Power Operated Relief Valves				
	FS-C	High	CS	CS (CSJ-03)	Fail Safe Test - Close
	PI	High	2Y	2Y	Position Indication
	ST-O	High	CS	CS (CSJ-03)	Stroke Time Measurement - Open
RC05	RCS PORV Block Valves				
	PI	High	2Y	Q	Position Indication
	ST-C	High	Q	Q	Stroke Time Measurement - Close
	ST-O	High	Q	Q	Stroke Time Measurement - Open
RC06	Reactor Vessel Head Vent Isolation Valves				
	FS-C	Low	CS	6YR	Fail Safe Test - Close
	PI	Low	2Y	6YR	Position Indication
	ST-C	Low	CS	6YR	Stroke Time Measurement - Close
	ST-O	Low	CS	6YR	Stroke Time Measurement - Open
RC07	Reactor Vessel Head Vent Throttle Valves				
	FS-C	Low	CS	3YR	Fail Safe Test - Close
	PI	Low	2Y	3YR	Position Indication

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
	ST-C	Low	CS	3YR	Stroke Time Measurement - Close
	ST-O	Low	CS	3YR	Stroke Time Measurement - Open
RC08	Pressurizer Relief Tank Vent to Gaseous Waste Processing System Inside Cntmt Isolation Valve (3652)				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
RC09	Reactor Make-up Water to RCP Standpipe and PRT Outside Containment Check Valve.				
	CV-C	Low	RF	APP J	Check Valve Close Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
RD01	RCS Vacuum Degassing from RCB ICIV and OCIV				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
RH01	Residual Heat Removal Heat Exchange Control Valve (Trains A, B, and C)				
	FS-O	Low	Q	54MO	Fail Safe Test - Open
	PI	Low	2Y	54MO	Position Indication
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
RH02	Residual Heat Removal Outlet to CVCS Letdown Valves				
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
	ST-O	Low	Q	3YR	Stroke Time Measurement - Open
RH03	Residual Heat Removal Pump Miniflow MOVs (Trains A, B, and C)				
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
RH04	Residual Heat Removal Inlet Isolation MOVs (Class 1 Boundary) Trains A, B, and C				
	LR-PIV	Medium	CS	54MO	Leak Rate Test - Pressure Isolation Valve
	PI	Medium	2Y	54MO	Position Indication
	ST-C	Medium	CS	54MO	Stroke Time Measurement - Close
	ST-O	Medium	CS	54MO	Stroke Time Measurement - Open
RH05	Residual Heat Removal Pump Miniflow Check Valves (Trains A, B, and C)				
	CV-O	Low	6M	54MO	Check Valve Open Exercise
RH06	Residual Heat Removal Pump Discharge Check Valves (Trains A, B, and C)				
	CV-O	Medium	CS	54MO	Check Valve Open Exercise
	CV-OP	Medium	6M	54MO	Check Valve Partial Open Exercise
RH07	Low Head Safety Injection to RCS Hot Leg Check Valves (Trains A, B, and C)				
	CV-C	Low	CS	54MO	Check Valve Close Exercise

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
	CV-O	Low	CS	54MO	Check Valve Open Exercise
	LR-PIV	Low	CS	54MO	Leak Rate Test - Pressure Isolation Valve
RH08	Cold Leg Injection Check Valves (Trains A, B, and C)				
	CV-C	Medium	CS	54MO	Check Valve Close Exercise
	CV-O	Medium	CS	54MO	Check Valve Open Exercise
	LR-PIV	Medium	CS	54MO	Leak Rate Test - Pressure Isolation Valve
RH09	RHR Return to RWST CIVs				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
RM01	Reactor Make-up Water Non-essential services isolation Valves				
	FS-C	Low	Q	3YR	Fail Safe Test - Close
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
SA01	Service Air to RCB Inside Cntmt Isolation Check Valve				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
SA02	Service Air to RCB Outside Cntmt Isolation Manual Valve				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
SB01	Steam Generator Bulk Water Sample Outside Cntmt Isolation Valves				
	FS-C	Low	Q	6YR	Fail Safe Test - Close
	PI	Low	2Y	6YR	Position Indication
	ST-C	Low	Q	6YR	Stroke Time Measurement - Close
SB02	Steam Generator Blowdown Outside Cntmt Isolation Valves				
	FS-C	Low	Q	6YR	Fail Safe Test - Close
	PI	Low	2Y	6YR	Position Indication
	ST-C	Low	Q	6YR	Stroke Time Measurement - Close
SD01	Starting Air Receiver Inlet Check Valves				
	CV-C	Low	Q	54MO	Check Valve Close Exercise
SI01	Safety Injection System Test Line Containment Isolation Valves				
	FS-C	Low	Q	3YR	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
SI02	Accumulator Nitrogen Supply Outside Cntmt Isolation Valve (3983)				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
SI03	Accumulator Nitrogen Supply Inside Cntmt Isolation Check Valve (SI0058)				
	CV-C	Low	RF	APP J	Check Valve Close Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
SI04	Reactor Water Storage Tank Clean-Up by SFPCCS Isolation Valves				
	FS-C	Low	Q	3YR	Fail Safe Test - Close
	PI	Low	2Y	3YR	Position Indication
	ST-C	Low	Q	3YR	Stroke Time Measurement - Close
SI05	Residual Heat Exchanger Bypass Valves (Trains A, B, and C)				
	FS-C	Low	CS	54MO	Fail Safe Test - Close
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	CS	54MO	Stroke Time Measurement - Close
SI06	Low Head Safety Injection Pump Discharge Outside Cntmt Isolation Valves (Trains A, B, and C)				
	LR-CIV-AL	Low	30 MO	54MO	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
SI07	Safety Injection Emergency Sump Outside Cntmt Isolation MOVs (Trains A, B, and C)				
	LR-CIV-AL	Medium	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Medium	2Y	54MO	Position Indication
	ST-C	Medium	Q	54MO	Stroke Time Measurement - Close
	ST-O	Medium	Q	54MO	Stroke Time Measurement - Open
SI08	High Head Safety Injection Pump Discharge Outside Cntmt Isolation Valves (Trains A, B, and C)				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
SI09	High Head Safety Injection Cold Leg Isolation (Trains A, B, and C)				
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
SI10	High Head Safety Injection Hot Leg Isolation (Trains A, B, and C)				
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
SI11	Residual Heat Removal Heat Exchanger Return to Hot Leg MOV (Trains A, B, and C)				
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
SI12	Cold Leg Injection MOVs (Trains A, B, C)				
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	Q	54MO	Stroke Time Measurement - Close
	ST-O	Low	Q	54MO	Stroke Time Measurement - Open
SI13	High Head Safety Injection Pump Recirc Isolation				
	PI	Medium	2Y	54MO	Position Indication
	ST-C	Medium	Q	54MO	Stroke Time Measurement - Close
	ST-O	Medium	Q	54MO	Stroke Time Measurement - Open
SI14	Low Head Safety Injection Pump Recirc Isolation				
	PI	Medium	2Y	54MO	Position Indication
	ST-C	Medium	Q	54MO	Stroke Time Measurement - Close
	ST-O	Medium	Q	54MO	Stroke Time Measurement - Open
SI15	Safety Injection System Reactor Water Storage Tank Isolation				
	PI	Medium	2Y	54MO	Position Indication
	ST-C	Medium	Q	54MO	Stroke Time Measurement - Close
	ST-O	Medium	Q	54MO	Stroke Time Measurement - Open
SI16	Accumulator Nitrogen Vent Valves (Trains A, B, and C)				
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	CS	54MO	Stroke Time Measurement - Close
	ST-O	Low	CS	54MO	Stroke Time Measurement - Open
SI17	Accumulator Nitrogen Vent Back-Up Valve (899)				
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	CS	R	Stroke Time Measurement - Close
	ST-O	Low	CS	R	Stroke Time Measurement - Open
SI18	High Head Safety Injection Pump Discharge Inside Cntmt Isolation Valves (Trains A, B, and C)				
	CV-C	Low	RF	APP J (VRR-03)	Check Valve Close Exercise
	CV-O	High	RF	R (ROJ-01)	Check Valve Open Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
SI19	High Head Safety Injection Pump Discharge Check to Cold Leg (Class 1 Boundary) (Trains A, B, and C)				
	CV-C	High	CS	CS (CSJ-04)	Check Valve Close Exercise
	CV-O	High	RF	R (ROJ-01)	Check Valve Open Exercise
	LR-PIV	High	CS	CS	Leak Rate Test - Pressure Isolation Valve
SI20	High Head Safety Injection Pump Discharge Check to Hot Leg (Class 1 Boundary) (Trains A, B, and C)				
	CV-C	Low	CS	54MO	Check Valve Close Exercise
	CV-O	Low	RF	54MO	Check Valve Open Exercise
	LR-PIV	Low	CS	54MO	Leak Rate Test - Pressure Isolation Valve

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
SI21	Low Head Safety Injection Pump Discharge Inside Cntmt Isolation Valves (Trains A, B, and C)				
	CV-C	Low	RF	APP J (VRR-03)	Check Valve Close Exercise
	CV-O	Medium	RF	54MO	Check Valve Open Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
SI22	Safety Injection System Pumps Discharge Check to Hot Leg (Class 1 Boundary) (Trains A, B, and C)				
	CV-C	Low	CS	54MO	Check Valve Close Exercise
	CV-O	Low	CS	54MO	Check Valve Open Exercise
	LR-PIV	Low	CS	54MO	Leak Rate Test - Pressure Isolation Valve
SI23	Accumulator to Cold Leg Inboard Check Valves (Trains A, B, and C)				
	CV-C	High	CS	CS (CSJ-04)	Check Valve Close Exercise
	CV-O	High	RF	R (ROJ-02)	Check Valve Open Exercise
	LR-PIV	High	CS	CS	Leak Rate Test - Pressure Isolation Valve
SI24	Accumulator Tank Discharge MOVs (Trains A, B, and C)				
	PI	Low	2Y	54MO	Position Indication
	ST-C	Low	CS	54MO	Stroke Time Measurement - Close
	ST-O	Low	CS	54MO	Stroke Time Measurement - Open
SI25	Safety Injection Pumps Suction Check Valves (Trains A, B, and C)				
	CV-OP	Medium	Q	54MO	Check Valve Partial Open Exercise
	DA	High	RF	54MO (ROJ-03)	Disassemble and Inspect
SI26	Accumulator Nitrogen Vent Header Bleed Valve (HCV-0900)				
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	CS	R	Stroke Time Measurement - Close
	ST-O	Low	CS	R	Stroke Time Measurement - Open
SI27	Accumulator to Cold Leg Outboard Check Valves (Trains A, B, and C)				
	CV-C	Low	CS	54MO	Check Valve Close Exercise
	CV-O	Low	RF	54MO	Check Valve Open Exercise
	LR-PIV	Low	CS	54MO	Leak Rate Test - Pressure Isolation Valve
SL1	High Pressure Sludge Lancing CIVs				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
SL2	Low Pressure Sludge Lancing CIVs				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
WL01	RCDT Vent Outside Containment Isolation Valve				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close

GROUP	Test	IST Rank	Frequency	RI-IST Frequency	IST TEST DESCRIPTION
WL02	RCDT To LWPS Outside Containment Isolation Valve				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
WL03	RCDT To LWPS Inside Containment Isolation Valve				
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
WL04	RCDT Vent Inside Containment Isolation Valve				
	FS-C	Low	Q	R	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	2Y	Position Indication
	ST-C	Low	Q	R	Stroke Time Measurement - Close
XC01	Reactor Containment Personal Air-lock Safety Check Valves (XC-48,49)				
	CV-C	Low	Q	3YR	Check Valve Close Exercise
	CV-O	Low	Q	3YR	Check Valve Open Exercise
	LR-CIV-AL	Low	30 MO	APP J	Leak Rate Test - Containment Isolation Valve
XC02	Reactor Containment Air-lock Air Supply Containment Isolation Valves (FV1025, 26,27,28)				
	FS-C	Low	Q	6YR	Fail Safe Test - Close
	LR-CIV-AL	Low	30 MO	6YR	Leak Rate Test - Containment Isolation Valve
	PI	Low	2Y	6YR	Position Indication
	ST-C	Low	Q	6YR	Stroke Time Measurement - Close

IST Pump Plan

GROUP								
System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
Pump Safety Function								

AFMDP Motor Driven AFW Pumps

System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
AF	3S141MPA02	5S141F00024	C-7	MOTOR DRIVEN AUX FEEDWATER PUMP NO. 12	3	High	Q	Q

The motor driven AFW pump is capable of delivering a minimum required feedwater flow of 540 gpm (UFSAR Section 6.2.1.4.5) to one steam generator during Loss Of Main Feedwater (w/wo offsite power available), Feedwater Line Break, Steam Line Break, Loss Of All AC Power, Control Room Evacuation, and Loss Of Coolant Accident events (DBD Section 3.2.8.9). The pump also functions to supply feedwater to one or more steam generators to perform cooldown of the Reactor Coolant System from normal zero load temperatures to a hot leg temperature of approximately 350F (DBD Section 3.2.1.5).

System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
AF	3S141MPA03	5S141F00024	B-7	MOTOR DRIVEN AUX FEEDWATER PUMP NO. 13	3	High	Q	Q

The motor driven AFW pump is capable of delivering a minimum required feedwater flow of 540 gpm (UFSAR Section 6.2.1.4.5) to one steam generator during Loss Of Main Feedwater (w/wo offsite power available), Feedwater Line Break, Steam Line Break, Loss Of All AC Power, Control Room Evacuation, and Loss Of Coolant Accident events (DBD Section 3.2.8.9). The pump also functions to supply feedwater to one or more steam generators to perform cooldown of the Reactor Coolant System from normal zero load temperatures to a hot leg temperature of approximately 350F (DBD Section 3.2.1.5).

System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
AF	3S141MPA01	5S141F00024	F-7	MOTOR DRIVEN AUX FEEDWATER PUMP NO. 11	3	High	Q	Q

The motor driven AFW pump is capable of delivering a minimum required feedwater flow of 540 gpm (UFSAR Section 6.2.1.4.5) to one steam generator during Loss Of Main Feedwater (w/wo offsite power available), Feedwater Line Break, Steam Line Break, Loss Of All AC Power, Control Room Evacuation, and Loss Of Coolant Accident events (DBD Section 3.2.8.9). The pump also functions to supply feedwater to one or more steam generators to perform cooldown of the Reactor Coolant System from normal zero load temperatures to a hot leg temperature of approximately 350F (DBD Section 3.2.1.5).

GROUP								
System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
Pump Safety Function								
AFTDP	Turbine Driven AFW Pump							
AF	3S141MPA04	5S141F00024	G-7	TURBINE DRIVEN AUX FEEDWATER PUMP	3	High	Q	Q

The turbine driven AFW pump is capable of delivering a minimum required feedwater flow of 540 gpm (UFSAR Section 6.2.1.4.5) to one steam generator during Loss Of Main Feedwater (w/wo offsite power available), Feedwater Line Break, Steam Line Break, Loss Of All AC Power, Control Room Evacuation, and Loss Of Coolant Accident events (DBD Section 3.2.8.9). The pump also functions to supply feedwater to one or more steam generators to perform cooldown of the Reactor Coolant System from normal zero load temperatures to a hot leg temperature of approximately 350F (DBD Section 3.2.1.5).

GROUP System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
Pump Safety Function								
CCPP	Component Cooling Water Pumps							
CC	3R201NPA101A	5R209F05017	B-7	COMPONENT COOLING WATER PUMP 1A	3	Medium	Q	54MO
Provides 14,070 gpm of cooling water (DBD 4.1.6.2) to ESF components under safe shutdown and accident conditions.								
CC	3R201NPA101B	5R209F05018	B-7	COMPONENT COOLING WATER PUMP 1B	3	Medium	Q	54MO
Provides 14,070 gpm of cooling water (DBD 4.1.6.2) to ESF components under safe shutdown and accident conditions.								
CC	3R201NPA101C	5R209F05019	B-7	COMPONENT COOLING WATER PUMP 1C	3	Medium	Q	54MO
Provides 14,070 gpm of cooling water (DBD 4.1.6.2) to ESF components under safe shutdown and accident conditions.								

GROUP								
System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
Pump Safety Function								

CHPP Chilled Water Pumps

System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
CH	3V111VPA004	5V119V10001	F-7	ESSENTIAL CHILL WATER PUMP 11A	3	Medium	Q	54MO

1.Provides the motive force for moving chilled water in a closed loop through the essentail chillers and cooling coils of the various safety related air handling units (AHUs).

2.Remain functional during and following all design basis accidents and plant safe shutdown.

NOTE:Receives an auto start signal upon SI initiation signal. Design flow is 981 gpm (per DBD).

System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
CH	3V111VPA005	5V119V10001	C-7	ESSENTIAL CHILL WATER PUMP 11B	3	Medium	Q	54MO

1. Provides the motive force for moving chilled water in a closed loop through the essential chillers and cooling coils of the various safety related air handling units (AHUs).

2. Remain functional during and following all design basis accidents and plant safe shutdown.

NOTE: Receives an auto start signal upon SI initiation signal. Design flow is 981 gpm (per DBD).

System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
CH	3V111VPA006	5V119V10001	A-7	ESSENTIAL CHILL WATER PUMP 11C	3	Medium	Q	54MO

1. Provides the motive force for moving chilled water in a closed loop through the essentail chillers and cooling coils of the various safety related air handling units (AHUs).

2. Remain functional during and following all design basis accidents and plant safe shutdown.

NOTE: Receives an auto start signal upon SI initiation signal. Design flow is 981 gpm (per DBD).

GROUP System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
Pump Safety Function								

CSPP Containment Spray pumps

CS	2N101NPA101A	5N109F05037	G-3	CONTAINMENT SPRAY PUMP 1A	2	Low	6M	54MO

1. Supply borated water from the Reactor Water Storage Tank to the Containment Spray ring header during the short-term injection phase upon receipt of a "HI-3" containment high pressure signal during a steam break inside containment or a LOCA to reduce containment pressure.

2. Recirculate borated water from the containment sumps to the Containment Spray header during the long-term recirculation phase subsequent to a main steam break inside of containment or a LOCA to reduce containment pressure.

CS	2N101NPA101B	5N109F05037	D-3	CONTAINMENT SPRAY PUMP 1B	2	Low	6M	54MO

1. Supply borated water from the Reactor Water Storage Tank to the Containment Spray ring header during the short-term injection phase upon receipt of a "HI-3" containment high pressure signal during a steam break inside containment or a LOCA to reduce containment pressure.

2. Recirculate borated water from the containment sumps to the Containment Spray header during the long-term recirculation phase ì□subsequent to a main steam break inside of containment or a LOCA to ì□reduce containment pressure.

CS	2N101NPA101C	5N109F05037	B-3	CONTAINMENT SPRAY PUMP 1C	2	Low	6M	54MO

1. Supply borated water from the Reactor Water Storage Tank to the Containment Spray ring header during the short-term injection phase upon receipt of a "HI-3" containment high pressure signal during a steam break inside containment or a LOCA to reduce containment pressure.

2. Recirculate borated water from the containment sumps to the Containment Spray header during the long-term recirculation phase subsequent to a main steam break inside of containment or a LOCA to reduce containment pressure.

GROUP								
System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
Pump Safety Function								
CVBAT	Boric Acid Transfer Pumps							
CV	3R171NPA103B	5R179F05009	C-4	BORIC ACID TRANSFER PUMP 1B	3	Low	Q	36MO
Transfer 110 gpm of boric acid solution from the boric acid tanks to the suction of the charging pumps during safety function boration operations (DBD 3.2.1.4).								
CV	3R171NPA103A	5R179F05009	D-4	BORIC ACID TRANSFER PUMP 1A	3	Low	Q	36MO
Transfer 110 gpm of boric acid solution from the boric acid tanks to the suction of the charging pumps during safety function boration operations (DBD 3.2.1.4).								

GROUP								
System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
Pump Safety Function								
CVCP	Centrifugal Charging Pump							
CV	2R171NPA101B	5R179F05007	D-5	CENTRIFUGAL CHARGING PUMP 1B	2	Medium	Q	36MO
Provide 112 gpm of boric acid solution to the Reactor Coolant System for boration through the charging flowpath and the seal injection flow path (DBD 3.2.2.1.4).								
CV	2R171NPA101A	5R179F05007	B-5	CENTRIFUGAL CHARGING PUMP 1A	2	Medium	Q	36MO
Provide 112 gpm of boric acid solution to the Reactor Coolant System for boration through the charging flowpath and the seal injection flow path (DBD 3.2.2.1.4).								

GROUP								
System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
Pump Safety Function								
EWPP	EW Pumps							
EW	3R281NPA101A	5N109F05038	C-3	ESSENTIAL COOLING WATER PUMP 1A	3	High	Q	Q

Takes a suction from the Emergency Cooling Pond and delivers cooling water to Emergency Diesel Generator heat exchangers, Essential Chillers, and Component Cooling Water heat exchangers during normal operating, shutdown, and following accident conditions. The ECW pumps receive an auto start signal upon an SI initiation signal.

Design Flow: 19,280 gpm (per DBD)

EW	3R281NPA101C	5N109F05038	C-3	ESSENTIAL COOLING WATER PUMP 1C	3	High	Q	Q

Takes a suction from the Emergency Cooling Pond and delivers cooling water to Emergency Diesel Generator heat exchangers, Essential Chillers, and Component Cooling Water heat exchangers during normal operating, shutdown, and following accident conditions. The ECW pumps receive an auto start signal upon an SI initiation signal.

Design Flow: 19,280 gpm (per DBD)

EW	3R281NPA101B	5N109F05038	C-3	ESSENTIAL COOLING WATER PUMP 1B	3	High	Q	Q

Takes a suction from the Emergency Cooling Pond and delivers cooling water to Emergency Diesel Generator heat exchangers, Essential Chillers, and Component Cooling Water heat exchangers during normal operating, shutdown, and following accident conditions. The ECW pumps receive an auto start signal upon an SI initiation signal.

Design Flow: 19,280 gpm (per DBD)

GROUP								
System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
Pump Safety Function								

EWSW ECW Screen Wash Pump

System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
EW	3R281NPA102A	5N109F05039	D-7	ECW SCREEN WASH BOOSTER PUMP 1A	3	Low	Q	54MO

The ECW Screen Wash Booster Pumps take water from the ECW pump discharge header and supply it to the ECW travelling screens at the required pressure and flow rate to clean the ECW travelling water screens. The pumps receive an auto start signal upon an SI initiation and will run continuously.

Design Flow: 176 gpm (per DBD)

System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
EW	3R281NPA102B	5N109F05039	D-4	ECW SCREEN WASH BOOSTER PUMP 1B	3	Low	Q	54MO

The ECW Screen Wash Booster Pumps take water from the ECW pump discharge header and supply it to the ECW travelling screens at the required pressure and flow rate to clean the ECW travelling water screens. The pumps receive an auto start signal upon an SI initiation and will run continuously.

Design Flow: 176 gpm (per DBD)

System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
EW	3R281NPA102C	5N109F05039	D-2	ECW SCREEN WASH BOOSTER PUMP 1C	3	Low	Q	54MO

The ECW Screen Wash Booster Pumps take water from the ECW pump discharge header and supply it to the ECW travelling screens at the required pressure and flow rate to clean the ECW travelling water screens. The pumps receive an auto start signal upon an SI initiation and will run continuously.

Design Flow: 176 gpm (per DBD)

GROUP								
System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
Pump Safety Function								
FCPP	Spent fuel pool cooling pumps							
FC	3R211NPA101A	5R219F05028	G-3	SPENT FUEL COOLING PUMP 1A	3	Low	Q	36MO
Circulates the spent fuel water through filter demineralizers to maintain purity and visual clarity of the spent fuel pool water, and through heat exchangers to remove the normal and maximum design heat load from the spent fuel pool. Design Flow: 2500 gpm (UFSAR Table 9.1-2)								
FC	3R211NPA101B	5R219F05028	D-3	SPENT FUEL COOLING PUMP 1B	3	Low	Q	36MO
Circulates the spent fuel water through filter demineralizers to maintain purity and visual clarity of the spent fuel pool water, and through heat exchangers to remove the normal and maximum design heat load from the spent fuel pool. Design Flow: 2500 gpm (UFSAR Table 9.1-2)								

GROUP								
System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
Pump Safety Function								
RHPP	RHR Pumps							
RH	2R161NPA101B	5R169F20000	E-6	RHR PUMP 1B	2	Medium	6M	54MO
Circulates 3000 gpm for final phase of reactor cooldown following a SBLOCA, SGTR, MSLB, FWLB, and in the event of a fire.								
RH	2R161NPA101C	5R169F20000	G-6	RHR PUMP 1C	2	Medium	6M	54MO
Circulates 3000 gpm for final phase of reactor cooldown following a SBLOCA, SGTR, MSLB, FWLB, and in the event of a fire.								
RH	2R161NPA101A	5R169F20000	B-6	RHR PUMP 1A	2	Medium	6M	54MO
Circulates 3000 gpm for final phase of reactor cooldown following a SBLOCA, SGTR, MSLB, FWLB, and in the event of a fire.								

GROUP								
System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
Pump Safety Function								

SIHHP High Head Safety Injection Pumps (Trains A, B, and C)

System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
SI	2N121NPA101A	5N129F05013	F-4	HIGH HEAD SAFETY INJECTION PUMP 1A	2	High	Q	Q

1. Inject borated water from the RWST to the RCS cold legs during the short term core cooling/cold-leg injection phase of safety injection. (Flow is required to be greater than 1470 gpm and less than 1620 gpm per T.S. Surveillance Requirement 4.5.2g.)

2. Recirculate borated water from the containment sump to the RCS cold or hot legs during the long term core cooling/cold and hotleg recirculation phase.

System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
SI	2N121NPA101B	5N129F05014	G-3	HIGH HEAD SAFETY INJECTION PUMP 1B	2	High	Q	Q

1. Inject borated water from the RWST to the RCS cold legs during the short term core cooling/cold-leg injection phase of safety injection. (Flow is required to be greater than 1470 gpm and less than 1620 gpm per T.S. Surveillance Requirement 4.5.2g.)

2. Recirculate borated water from the containment sump to the RCS cold or hot legs during the long term core cooling/cold and hotleg recirculation phase.

System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
SI	2N121NPA101C	5N129F05015	F-3	HIGH HEAD SAFETY INJECTION PUMP 1C	2	High	Q	Q

1. Inject borated water from the RWST to the RCS cold legs during the short term core cooling/cold-leg injection phase of safety injection. (Flow is required to be greater than 1470 gpm and less than 1620 gpm per T.S. Surveillance Requirement 4.5.2g.)

2. Recirculate borated water from the containment sump to the RCS cold or hot legs during the long term core cooling/cold and hotleg recirculation phase.

GROUP System	PUMP Tag No.	PID Drawing No.	Coord.	PUMP NAME	CLASS	IST Rank	Frequency	RI-IST Freq.
Pump Safety Function								
SILHP	Low Head Safety Injection Pumps (Trains A, B, and C)							
SI	2N121NPA102C	5N129F05015	C-3	LOW HEAD SAFETY INJECTION PUMP 1C	2	Medium	Q	54MO

1. Inject borated water from the RWST to the RCS cold legs during the short term core cooling/cold-leg injection phase of safety injection. (Flow is required to be greater than 2550 gpm and less than 2800 gpm per T.S. Surveillance Requirement 4.5.2g.)

2. Recirculate borated water from the containment sump to the RCS cold or hot legs during the long term core cooling/cold and hotleg recirculation phase.

SI	2N121NPA102A	5N129F05013	C-3	LOW HEAD SAFETY INJECTION PUMP 1A	2	Medium	Q	54MO

1.Inject borated water from the RWST to the RCS cold legs during the short term core cooling/cold-leg injection phase of safety injection. (Flow is required to be greater than 2550 gpm and less than 2800 gpm per T.S. Surveillance Requirement 4.5.2g.)

2. Recirculate borated water from the containment sump to the RCS cold or hot legs during the long term core cooling/cold and hotleg recirculation phase.

SI	2N121NPA102B	5N129F05014	D-3	LOW HEAD SAFETY INJECTION PUMP 1B	2	Medium	Q	54MO

1. Inject borated water from the RWST to the RCS cold legs during the short term core cooling/cold-leg injection phase of safety injection. (Flow is required to be greater than 2550 gpm and less than 2800 gpm per T.S. Surveillance Requirement 4.5.2g.)

2. Recirculate borated water from the containment sump to the RCS cold or hot legs during the long term core cooling/cold and hotleg recirculation phase.

Code Testing Exceptions Report

Test Exception Number	*Test Exception Type*

CSJ-01

Cold Shutdown Justification

Group AF02 — Auxiliary Feedwater Supply to Steam Generator Outside Cntmt Isolation Stop Check MOVs

Safety Function

1. Open upon receipt of:
A. steam generator low water level,
B. low feedwater flow signal from AMSAC, or
C. SI initiation signal to allow 500 gpm (per Technical Specification 4.7.1.2.1) flow to Steam Generator 1(2)D.

NOTE: The ESF actuation signal allows the stop check valve to function normally through the self-actuating design of the check valve. Operation of the motor operator function is not required for the valve to fulfill its open safety function.

2. Close (remote manual) for Steam Generator 1(2)D isolation in response to SGTR, FWLB, and MSLB.

Code Required Tests OMa 4.3.2.1 requires check valves to be exercised nominally every three (3) months. OMa 4.3.2.2 requires that each check valve be exercised or examined in a manner that verifies obturator travel to the closed, full-open or partially open position required to fullfill its safety function.

Reason for Exception These valves can only be full-stroke open exercised by directing auxiliary feedwater flow into the steam generator. The initiation of auxiliary feedwater flow during power operation would result in unwanted thermal shock to the secondary portions of the steam generators. Additionally, the introduction of cold water to the steam generator would cause an unwanted power transient.

Alternate Testing These valves will be full-stroke open exercised each cold shutdown unless the period of time since the previous full-stroke open exercise is less than three months. Auxiliary feedwater flow will be directed through the valve from its respective pump and into the steam generator. Verification of flow through the valve will provide assurance that the valve has opened sufficiently to perform its safety function.

Test Exception Number	*Test Exception Type*
CSJ-02	Cold Shutdown Justification

Group AF01 — Auxiliary Feedwater Supply to Steam Generator Inside Cntmt Isolation Check Valves

Safety Function 1. Open to allow 500 gpm (per Technical Specification 4.7.1.2.1) of auxiliary feedwater flow to Steam Generator 1(2)A.

Code Required Tests OMa 4.3.2.1 requires check valves to be exercised nominally every three (3) months. OMa 4.3.2.2 requires that each check valve be exercised or examined in a manner that verifies obturator travel to the closed, full-open or partially open position required to fullfill its safety function.

Reason for Exception These valves can only be full-stroke open exercised by directing auxiliary feedwater flow into the steam generator. The initiation of auxiliary feedwater flow during power operation would result in unwanted thermal shock to the secondary portions of the steam generators. Additionally, the introduction of cold water to the steam generator would cause an unwanted power transient. Main feedwater flow cannot be used to exercise this check valve during normal power operation due to the thermal shock that would occur by injecting the cooler, stagnant water in the connecting piping. Flow instrumentation is not available in this configuration to verify that the valve has been properly exercised.

Alternate Testing These valves will be full-stroke open exercised each cold shutdown unless the period of time since the previous full-stroke open exercise is less than three months. Auxiliary feedwater flow will be directed through the valve from its respective pump and into the steam generator. Verification of flow through the valve will provide assurance that the valve has opened sufficiently to perform its safety function.

Test Exception Number	*Test Exception Type*
CSJ-03	Cold Shutdown Justification

Group RC04 RCS Power Operated Relief Valves

Safety Function 1. Remain closed to preserve the integrity of the reactor coolant pressure boundary.

2. Open to depressurize the RCS to cold shutdown conditions and to mitigate transients/accidents such as MSLB and FWLB.

3. Open during the long term cooling mode following a SBLOCA to satisfy LHSI pump minimum flow requirements.

4. Open in response to COMS to provide overpressure mitigation for the RCS and prevent pressure-temperature conditions from exceeding Appendix G limits.

Code Required Tests OMa 4.2.1.1 requires that each active Category B valve be tested nominally every three (3) months for operational readiness.

Reason for Exception The operability testing (full-stroke open and close exercise) of these valves during normal power operation would require closing the associated block valve to prevent an undesirable RCS pressure and pressurizer level transients. Failure of the valve to properly reseat after the open and close exercise test would require the block valve to be closed and entry into a Limiting Condition for Operation with a possible plant shutdown being required.

Alternate Testing These valves will be full-stroke open and close exercised, stroke timed, and their fail-safe actuation verified at each cold shutdown not to exceed once every three months per the requirements of OMa 4.2.1.2

Test Exception Number	*Test Exception Type*
CSJ-04	Cold Shutdown Justification

Group SI19 High Head Safety Injection Pump Discharge Check to Cold Leg (Class 1 Boundary) (Trains A, B, and C)

Safety Function

1. Open to inject borated water from either the RWST or the containment sump to the RCS cold legs during the cold leg injection phase of safety injection (Flow rate required is >1,470 gpm and <1620 gpm for HHSI pump lines following completion of modifications to the system that alters its flow characteristics per Technical Specification 4.5.2.g).

2. Close to prevent the diversion of flow from the accumulator or from the LHSI pump in the event that the corresponding HHSI pump is not running.

3. Close and be leak tight (CAT A) to maintain RCS pressure boundary, GDC 14 (PIV).

Code Required Tests Oma 4.3.2.1 requires that each active Category A/C valve be tested nomilnally every three (3) months.

Reason for Exception The close exercise testing of these valves will be in conjucntion with the seat leakage testing require by Oma 4.2.2.3. The seat leakage testing must be performed with the maximum differential pressure across the valve seats. In addition, the following normally de-energized valves must be energized and remain energized in the abnormal valve position until testing is completed and the valves are returned to their normal operating position.
2N121(2) XSI0039A,B,C - Accumulator Tank Discharge Isolation Valves.
2N121(2)XSI0008A,B,C - HHSI Lot Leg Isolation Valves
2R161(2)XRH0019A, B, C - RHR Heat Exchanger Return to Hot Leg Valves
2R161(2)XRH0031A,B,C - Cold Leg Injection Valves

Alternate Testing These valves will be close exercised tested by the performance of a seat leakage test following each cold shutdown and prior to entering Mode 2 not to exceed once every nine months per the requirements of Technical Specification 4.4.6.2.2

Test Exception Number	*Test Exception Type*
CSJ-04	Cold Shutdown Justification

Group SI23 — Accumulator to Cold Leg Inboard Check Valves (Trains A, B, and C)

Safety Function

1. Open when the RCS pressure falls below the accumulator pressure to force borated water into the RCS cold legs.

2. Close to prevent backflow from the RCS into the low pressure SI system.

3. Close and be leak tight (CAT A) to maintain RCS pressure boundary, GDC 14 (PIV).

Code Required Tests Oma 4.3.2.1 requires that each active Category A/C valve be tested nomilnally every three (3) months.

Reason for Exception The close exercise testing of these valves will be in conjucntion with the seat leakage testing require by Oma 4.2.2.3. The seat leakage testing must be performed with the maximum differential pressure across the valve seats. In addition, the following normally de-energized valves must be energized and remain energized in the abnormal valve position until testing is completed and the valves are returned to their normal operating position.
2N121(2) XSI0039A,B,C - Accumulator Tank Discharge Isolation Valves.
2N121(2)XSI0008A,B,C - HHSI Lot Leg Isolation Valves
2R161(2)XRH0019A, B, C - RHR Heat Exchanger Return to Hot Leg Valves
2R161(2)XRH0031A,B,C - Cold Leg Injection Valves

Alternate Testing These valves will be close exercised tested by the performance of a seat leakage test following each cold shutdown and prior to entering Mode 2 not to exceed once every nine months per the requirements of Technical Specification 4.4.6.2.2

PRR-01 Pump Relief Request

Group EWPP EW Pumps

Safety Function Takes a suction from the Emergency Cooling Pond and delivers cooling water to Emergency Diesel Generator heat exchangers, Essential Chillers, and Component Cooling Water heat exchangers during normal operating, shutdown, and following accident conditions. The ECW pumps receive an auto start signal upon an SI initiation signal.

Design Flow: 19,280 gpm (per DBD)

Code Required Tests OMa Part 6, 5.2.1(b) requires the system resistance to be varied until the flow rate equals the reference point. The differential pressure shall be determined and compared to its reference value. Alternatively, the flow rate shall be varied until the differential pressure equals the reference point and the flow rate determined and compared to its reference value.

OMa Part 6, 5.2.1(c) states that where system resistance cannot be varied, flow rate and pressure shall be determined and compared to their respective reference values.

Reason for Exception The Essential Cooling Water System is designed so that total pump flow cannot be readily adjusted to one reference value for testing without adversely affecting the operating system flow balance or utilizing excessive operator resources which would be better utilized to monitor the safe operation of the plant. These pumps must be tested in a manner that does not adversely affect the flow balance and system operability.

System resistance is not fixed since each load has an acceptable flow range. Adjusting system total flow to meet a specific reference value may change the individual load flow rates and may cause one or more of the loads to move outside its respective operation range possibly requiring an entry into an LCO. Additionally, STP has specific "cold" and "warm" weather lineups for operation of the essential chillers creating a different system resistance. Consequently, adjusting flow to one specific value on a quarterly basis for the performance of pump testing conflicts with system design and challenges the system operability.

Alternate Testing As an alternative to the testing requirments of OMa Part 6, 5.2.1, STP will assess pump performance and operational readiness through the use of reference pump curves. Flow rate and pump differential pressure will be measured during inservice testing in the as found condition of the system and compared to an established reference curve. The following elements will be used in the development of the reference pump curves:

1. A reference pump curve (flow rate versus differential pressure) will be established for each of the ECW pumps for the data taken when these pumps are known to be operating acceptably.

2. Pump curves will be established from measurements taken with instrumentation

meeting or exceeding the accuracy requirements of OMa Part 6, 4.6.1.1.

3. Each Pump curve will be based on at least 5 points beyond the flat portion of the pump curve in the normal operating range of the pumps (at a flow greater than 15,700 gpm). Rated capacity of these pumps is 19,280 gpm. The pumps will be tested over the range of their full design flow rates, 15,700 gpm minumum to 20,610 gpm maximum.

4. The reference pump curves will be based on flow rate versus differential pressure. The acceptance criteria (acceptable and required action ranges) curves will be based on the differential pressure limits of OMa Part 6, Table 3b.

5. Vibration levels will be measured at each of the reference points. If negligible variation readings are observed over the range of pump conditions, a single reference value may be assigned to each vibration measurement location. If vibration levels change over the range of pump conditions, appropriate acceptance criteria will be asigned to regions of the pump curve.

6. After any maintenance or repair that may affect the existing reference pump curve, a new reference curve shall be determined or the existing pump curve revalidated by an inservice test. A new pump curve shall be established based on at least 5 points beyond the flat portion of the pump curve.

Test Exception Number	*Test Exception Type*
PRR-02	Pump Relief Request

Group CCPP — Component Cooling Water Pumps

Safety Function

Code Required Tests OMa-1988 Part 6, Paragraphs 4.6.1.1 and 4.6.1.2 require pressure instrumentation requirements for accuracy and range. Accuracy must be +/- 2% and full-scale range shall be not greater than three times the reference value.

Reason for Exception The installed suction pressure gauges for the Component Cooling Water pumps have a range of 160 psig and an accuracy of 0.5%. The reference values for suction pressure for these pumps have been as low as 21 psig. The installed suction pressure gauges fo rthe Component Cooling Water pumps have a full-scale range greater than 3 times the reference value, but have an accuracy of +/- 0.5%, which is more conservative than the Code. The combination of the range and accuracy of the installed suction pressure gauge yields a reading at least equivalent to the reading achieved from instruments that meet the Code Requirements. The installed suction pressure gauge meets the intent of the Code requirements and provides for an acceptable level of quality and safety for inservice testing.

Alternate Testing The permanently installed suction gauges for Component Cooling Water pumps 1A(2A), 1B(2B), and 1C(2C) will be used to obtain test measurements for evaluating pump operability.

Test Exception Number	Test Exception Type
ROJ-01	Refueling Outage Justification

Group SI19 — High Head Safety Injection Pump Discharge Check to Cold Leg (Class 1 Boundary) (Trains A, B, and C)

Safety Function

1. Open to inject borated water from either the RWST or the containment sump to the RCS cold legs during the cold leg injection phase of safety injection (Flow rate required is >1,470 gpm and <1620 gpm for HHSI pump lines following completion of modifications to the system that alters its flow characteristics per Technical Specification 4.5.2.g).

2. Close to prevent the diversion of flow from the accumulator or from the LHSI pump in the event that the corresponding HHSI pump is not running.

3. Close and be leak tight (CAT A) to maintain RCS pressure boundary, GDC 14 (PIV).

Code Required Tests OMa 4.3.2.1 requires check valves to be exercised nominally every three (3) months. OMa 4.3.2.2 requires that each check valve be exercised or examined in a manner that verifies obturator travel to the closed, full-open, or partially open position required to fullfill its safety function.

Reason for Exception These check valves cannot be exercised during normal power operation since the HHSI pump cannot overcome normal RCS pressure. These valves cannot be exercised at cold shutdown due to the possibility of over pressurizing the Reactor Coolant System.

Alternate Testing Per Oma 4.3.2.2.e, these check valves will be exercised, full stroke open, eachrefueling outage by injecting HHSI flow into the open RCS with a vent path established.

The most practical method of verifiying valve closure on cessation of flow or flow reversal is in conjucntion with the leakage testng require by technical specifications.

Valves 1N121(2)XSI0007A,B,C and 1N121(2)XSI0009A,B,C will be closed exercised tested in accordance with CSJ-04.

Valves 2N121(2)XSI0005A,B,C and 2N121(2)XSI0030A,B,C will be closed exercised tested in accordance with VRR-03.

Test Exception Number	Test Exception Type
ROJ-01	Refueling Outage Justification

Group SI18 — High Head Safety Injection Pump Discharge Inside Cntmt Isolation Valves (Trains A, B, and C)

Safety Function

1. Open to inject borated water from either the RWST or the containment sump to the RCS cold legs during the cold leg injection phase of safety injection (Flow rate required is >1,470 gpm and <1620 gpm for HHSI pump lines following completion of modifications to the system that alters its flow characteristics per Technical Specification 4.5.2.g).

2. Open to recirculate borated water from the containment sump to the RCS hot legs during the hot leg recirculation phase of safety injection.

3. Close and be leak tight (CAT A) to provide containment integrity.

Code Required Tests OMa 4.3.2.1 requires check valves to be exercised nominally every three (3) months. OMa 4.3.2.2 requires that each check valve be exercised or examined in a manner that verifies obturator travel to the closed, full-open, or partially open position required to fullfill its safety function.

Reason for Exception These check valves cannot be exercised during normal power operation since the HHSI pump cannot overcome normal RCS pressure. These valves cannot be exercised at cold shutdown due to the possibility of over pressurizing the Reactor Coolant System.

Alternate Testing Per Oma 4.3.2.2.e, these check valves will be exercised, full stroke open, each refueling outage by injecting HHSI flow into the open RCS with a vent path established.

The most practical method of verifiying valve closure on cessation of flow or flow reversal is in conjucntion with the leakage testng require by technical specifications.

Valves 1N121(2)XSI0007A,B,C and 1N121(2)XSI0009A,B,C will be closed exercised tested in accordance with CSJ-04.

Valves 2N121(2)XSI0005A,B,C and 2N121(2)XSI0030A,B,C will be closed exercised tested in accordance with VRR-03.

Test Exception Number	*Test Exception Type*
ROJ-02	Refueling Outage Justification

Group SI23 — Accumulator to Cold Leg Inboard Check Valves (Trains A, B, and C)

Safety Function 1. Open when the RCS pressure falls below the accumulator pressure to force borated water into the RCS cold legs.

2. Close to prevent backflow from the RCS into the low pressure SI system.

3. Close and be leak tight (CAT A) to maintain RCS pressure boundary, GDC 14 (PIV).

Code Required Tests OMa 4.3.2.1 requires check valves to be exercised nominallly every three (3) months. OMa 4.3.2.2 requires that each check valve be exercised or examined in a manner that verifies obturator travel to the closed, full-open, or partially open position required to fullfill its safety function.

Reason for Exception These check valves cannot be exercised during normal pewer operation (full or partial stroke open) since neither the HHSI, LHSI, RHR pump, or Accumulators can over come normall RCS pressure. These valves cannot be exercised at cold shutdown due to the possibility of over pressurizing the RCS.

Alternate Testing Per OMa 4.3.2.2.e, these check valves will be exercised, full stroke open, each refueling outage using non-intrusive techniques to ensure no degradation has occurred. If any check valve tested during the refueling outage shows signs of unacceptable degradation or performance, it will be disassembled and inspected during that refueling outage.

Test Exception Number	*Test Exception Type*

ROJ-03 Refueling Outage Justification

Group SI25 Safety Injection Pumps Suction Check Valves (Trains A, B, and C)

Safety Function 1. Open to provide a source of borated water to the suction of the LHSI, HHSI and CS pumps during the injection mode of accident mitigation (Flow rate required is 5920 gpm. This is a combination of 1470 gpm for HHSI, 2550 gpm for LHSI, and 1900 gpm for CS).

2. Close to prevent backflow to the RWST when containment sump isolation valves are opened during switchover from the injection phase to the cold leg recirculation mode before SI-MOV001A, B, and C are closed. Operator action is required to manually close SI-MOV001A, B, and C to complete the switchover process.

Code Required Tests OMa 4.3.2.1 requires check valves to be exercised nominallly every three (3) months. OMa 4.3.2.2 requires that each check valve be exercised or examined in a manner that verifies obturator travel to the closed, full-open, or partially open position required to fullfill its safety function.

Reason for Exception These check valves can only be exercised, full stroke, by simulating LOCA conditions and allowing the above pumps to inject flow into the RCS at zero or a very low pressure. These conditions can only be simulated during a refueling outage with the reactor vessel head off and the containment spray pump on full recirculation.

Closure of these check valves cannot be verified by non-intrusive means. There are no external position indicators on these valves and due to the soft closure of these valves (result of pump coastdown) acoustic methods are not conclusive. Magnetic methods are also not conclusive.

Draindown of a portion of the safety injection system is required t perform disassembly and inspection of the valves. Disassembly and inspection can only be accomplished during the 7 day Safety Injection System LCO window or during refueling outages.

Local leakage rate testing of other SI valves and other miantenance activiites are now being conducted during the 7 day SI system LCO windows. Conducting the disassembly and inspection of these check valves in conjunction with LLRTs or other maintenance activities would accomplish the following:
a) Increase the availability of the Safety Injection System during refueling outages which would lower the overall risk during the outages. The online risk should not be increased if performed during the AOT window since the SI Train will already be removed from service for LLRTs or other maintenance.
b) Radwaste should be reduced as the inspections will be performed with other draindown work during the LCO week.
c) There will be a reduction in outage manpower and resource requirements for

both maintenance and operations personnel.
d) A reduction in radiation exposure should be realized because personnel will have to perform drain and fill operations only once.

Alternate Testing Per OMa 4.3.2.2.e, these check valves will be exercised, full stroke, each refueling outage by injecting flow into the RCS with the vessel head off and the CS pump on full recirculation.

For closure verfication: Per OMa 4.3.2.4.c, if other test methods are impractical, a sample disassembly examination program shall be used to verfiy valve obturator movement. At least one check valve from the sample group will be verified operable by disassembly and inspection on a nominal refueling cycle frequency of 18 months (+/- 25%). This will not result in a reduction in the number of inspections performed over the life of the plant. If a generic failure occurs, a plan of action for inspection the remaining valves will be developed utilizing the Condition Reporting Process and the guidance provided in Generic Letter 91-18. This plan of action will take into account the potential failure modes and their associated plant impacts and will be implemented in a time frame commensurate with their safety significance. This will ensure that all check valves in this sample group are inspeced within six years as required by Generic Letter 89-04, Position 2. Appproval of this Relief Request will not preclude STP from performing these inspections during refueling outage should some other scope of work make it necessary to drain a train of SI.

Test Exception Number	Test Exception Type
VRR-01	Valve Relief Request
Group AP01	RCS Hot Leg Sample to PASS Lab OCIVs
Safety Function	1. Close in response to an ESF signal and leak tight (CAT A) to maintain containment integrity.
Code Required Tests	OMa 4.1 requires that valves with remote position indicators be observed locally at least once every two years to verify that valve operation is accurately indicated.
Reason for Exception	These valves are solenoid valves for which stem movement cannot be directly observed. They are redundant valves in series and operate simultaneously from a single control switch with one set of indicating lights.
Alternate Testing	These valves are stroked and timed during normal inservice testing using the remote indicating lights. Open and closed indicatin is actuated by the limit switches of each valve wired in series and remote postion indicatin is based on the slowest valve. Since these redundant valves cannot be exercised separately (unless leads are lifted, temporary power supplied to the disabled valve to hold it in the open position, and jumpers placed across the disabled valve's limit switches) the valves will be stroked simultaneoulsy and remote position indication verified by observing that system flow is initiated and then secured.

Test Exception Number	Test Exception Type
VRR-02	Valve Relief Request

Group AP01 — RCS Hot Leg Sample to PASS Lab OCIVs

Safety Function 1. Close in response to an ESF signal and leak tight (CAT A) to maintain containment integrity.

Code Required Tests Oma 4.2.1.1 requires that each category A valve be tested nominally every three months for operational readiness.

Reason for Exception The valves are redundant valves in series and operate simultaneously from a single control switch with one set of indicating lights. These redundant valves cannot be exercised separately (unless leads are lifted, temporary power supplied to the disabled valve to hold it in the open position, and jumpers placed across the disabled valve's limit switches).

Based on the guidance on NUREG 1482, an evaluation was performed and it was determined that only one valve is required to satisfy the plant safety analysis. Both valves will be included in the IST plan.

Alternate Testing Since these redundant valves cannot be exercised separately, the valves will be stroked simultaneously and timed using the remote position indication of the slowest valve. Failure to meet the stroke time acceptance criteria of OMa 4.2.1.8 shall be treated as a failure of a series valve pair and corrective actions taken to determine the cause of the failure.

Test Exception Number	*Test Exception Type*

VRR-03 Valve Relief Request

Group SI18 High Head Safety Injection Pump Discharge Inside Cntmt Isolation Valves (Trains A, B, and C)

Safety Function

1. Open to inject borated water from either the RWST or the containment sump to the RCS cold legs during the cold leg injection phase of safety injection (Flow rate required is >1,470 gpm and <1620 gpm for HHSI pump lines following completion of modifications to the system that alters its flow characteristics per Technical Specification 4.5.2.g).

2. Open to recirculate borated water from the containment sump to the RCS hot legs during the hot leg recirculation phase of safety injection.

3. Close and be leak tight (CAT A) to provide containment integrity.

Code Required Tests OMa 4.3.2.1 requires check valves to be exercised nominally every three (3) months. OMa 4.3.2.2 requires that each check valve be exercised or examined in a manner that verifies obturator travel to the closed, full-open, or partially open position required to fullfill its safety function.

Reason for Exception These check valves have a safety function in the closed direction as containment isolation valves. There are no intra or intersystem cross-ties downstream of these valves which would cause a diversion of flow from another pump if the check valve did not close. Due to the fact that there are no cross-ties downstream of the valves, the valves lack design provisions for system testing to verify closure capability in any plant condition.

Leak rate testing verifies valve closure by validating the valve seats properly and is leak tight, and provides more information about the closed position than a simple backflow test.

NUREG 1482, Sectin 4.1.4, allows the extension of the test interval to refueling outage frequency for check valves where the only practical means of verifiying check valve closure is by performing the Appendix J Leak Test. STP has adopted Option B of Appendix J that allows these check valves to be leak tested on a frequency not to exceed once every five years.

Disassembly provides limiited information on a check valve's ability to seat properly on cessation of flow. Following reassembly, the Code requires a post-assembly test which would reopen the check valve without providing assurance the disk would return to the closed position. Disassembly of these check valves is not practical due to the design complexity of the check valves, the increased probability of human error during valve reassembly, foreign material exclusion concerns, and ALARA considerations.

The subject valves have exhibited a history of satisfactory operation. Based on their performance history, it is believed that the current Probabilistic Risk

Assessment (PRA) modeling of the failure rates for these valves is still accurate. Irrespective of the failure rate modeling, the current STPNOC PRA model indicates that the potential failure of these valves to close has no impact on core damage frequency. In addition, the impact on these valves (assuming complete failure) from a Large Early Release standpoint in minimal.

Based on the above, it is evident that in the event that containment isolation is necessary, the subject valves will have a high probability of performing their intended safety function. Therefore, STP believes that the safety significance and potential consequences of the proposed relief is extremely small.

Alternate Testing Closure verification of these check valves will be performed by leak rate testing in accordance with 10CFR50 Appendix J on a frequency specified by Option B of Appendix J.

Test Exception Number	*Test Exception Type*
VRR-03	Valve Relief Request

Group SI21 — Low Head Safety Injection Pump Discharge Inside Cntmt Isolation Valves (Trains A, B, and C)

Safety Function

1. Open to inject borated water from either the RWST or the containment sump to the RCS cold legs during the cold leg injection phase of safety injection (Flow rate required is >2550 gpm and <2800 gpm for LHSI pump lines following completion of modifications to the system that alters its flow characteristics per Technical Specification 4.5.2.g).

2. Open to recirculate borated water from the containment sump to the RCS hot legs during the hot leg recirculation phase of safety injection.

3. Close to prevent backflow from the RHR system during post accident recovery operations.

4. Close and be leak tight (CAT A) to maintain containment integrity.

Code Required Tests

OMa 4.3.2.1 requires check valves to be exercised nominally every three (3) months. OMa 4.3.2.2 requires that each check valve be exercised or examined in a manner that verifies obturator travel to the closed, full-open, or partially open position required to fullfill its safety function.

Reason for Exception

These check valves have a safety function in the closed direction as containment isolation valves. There are no intra or intersystem cross-ties downstream of these valves which would cause a diversion of flow from another pump if the check valve did not close. Due to the fact that there are no cross-ties downstream of the valves, the valves lack design provisions for system testing to verify closure capability in any plant condition.

Leak rate testing verifies valve closure by validating the valve seats properly and is leak tight, and provides more information about the closed position than a simple backflow test.

NUREG 1482, Sectin 4.1.4, allows the extension of the test interval to refueling outage frequency for check valves where the only practical means of verifiying check valve closure is by performing the Appendix J Leak Test. STP has adopted Option B of Appendix J that allows these check valves to be leak tested on a frequency not to exceed once every five years.

Disassembly provides limiited information on a check valve's ability to seat properly on cessation of flow. Following reassembly, the Code requires a post-assembly test which would reopen the check valve without providing assurance the disk would return to the closed position. Disassembly of these check valves is not practical due to the design complexity of the check valves, the increased probability of human error during valve reassembly, foreign material exclusion concerns, and ALARA considerations.

The subject valves have exhibited a history of satisfactory operation. Based on their performance history, it is believed that the current Probabilistic Risk Assessment (PRA) modeling of the failure rates for these valves is still accurate. Irrespective of the failure rate modeling, the current STPNOC PRA model indicates that the potential failure of these valves to close has no impact on core damage frequency. In addition, the impact on these valves (assuming complete failure) from a Large Early Release standpoint in minimal.

Based on the above, it is evident that in the event that containment isolation is necessary, the subject valves will have a high probability of performing their intended safety function. Therefore, STP believes that the safety significance and potential consequences of the proposed relief is extremely small.

Alternate Testing Closure verification of these check valves will be performed by leak rate testing in accordance with 10CFR50 Appendix J on a frequency specified by Option B of Appendix J.

Test Exception Number	*Test Exception Type*

VRR-03 Valve Relief Request

Group CC29 — CCW Supply to RHR Pump and Heat Exchanger Inside Cntmt Isolation Check Valve (Trains A, B, and C)

Safety Function

1. Open to provide flow path for CCW through RHR pump 1(2)C seal cooler and RHR 1(2)C heat exchanger (4906 gpm required per DBD Table T-7, Minimum or Maximum Safeguards).

2. Close and leak tight (CAT A) in accordance with UFSAR commitment (Section 6.2.6.3 and Figure 6.2.4-1, Sheet 39) to provide containment integrity.

Code Required Tests

OMa 4.3.2.1 requires check valves to be exercised nominally every three (3) months. OMa 4.3.2.2 requires that each check valve be exercised or examined in a manner that verifies obturator travel to the closed, full-open, or partially open position required to fullfill its safety function.

Reason for Exception

These check valves have a safety function in the closed direction as containment isolation valves. There are no intra or intersystem cross-ties downstream of these valves which would cause a diversion of flow from another pump if the check valve did not close. Due to the fact that there are no cross-ties downstream of the valves, the valves lack design provisions for system testing to verify closure capability in any plant condition.

Leak rate testing verifies valve closure by validating the valve seats properly and is leak tight, and provides more information about the closed position than a simple backflow test.

NUREG 1482, Sectin 4.1.4, allows the extension of the test interval to refueling outage frequency for check valves where the only practical means of verifiying check valve closure is by performing the Appendix J Leak Test. STP has adopted Option B of Appendix J that allows these check valves to be leak tested on a frequency not to exceed once every five years.

Disassembly provides limiited information on a check valve's ability to seat properly on cessation of flow. Following reassembly, the Code requires a post-assembly test which would reopen the check valve without providing assurance the disk would return to the closed position. Disassembly of these check valves is not practical due to the design complexity of the check valves, the increased probability of human error during valve reassembly, foreign material exclusion concerns, and ALARA considerations.

The subject valves have exhibited a history of satisfactory operation. Based on their performance history, it is believed that the current Probabilistic Risk Assessment (PRA) modeling of the failure rates for these valves is still accurate. Irrespective of the failure rate modeling, the current STPNOC PRA model indicates that the potential failure of these valves to close has no impact on core damage frequency. In addition, the impact on these valves (assuming complete failure) from a Large Early Release standpoint in minimal.

Based on the above, it is evident that in the event that containment isolation is necessary, the subject valves will have a high probability of performing their intended safety function. Therefore, STP believes that the safety significance and potential consequences of the proposed relief is extremely small.

Alternate Testing Closure verification of these check valves will be performed by leak rate testing in accordance with 10CFR50 Appendix J on a frequency specified by Option B of Appendix J.