Exelon Generation

James A. FitzPatrick NPP
P.O. Box 110
Lycoming, NY 13093

William C. Drews
Regulatory Assurance Manager

JAFP-18-0078
August 6, 2018

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

James A. FitzPatrick Nuclear Power Plant
Renewed Facility Operating License No. DPR-59
NRC Docket No. 50-333

Subject: Fifth Ten-Year Interval Inservice Testing Program Plan

Dear Sir or Madam:

The Fifth 10-Year Inservice Testing (IST) Interval began at James A. FitzPatrick Nuclear Power Plant on June 1, 2018. The Enclosure to this letter contains the IST program plan for your records; submitted in accordance with American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code) 2004, subsection ISTA-3200(a).

There are no new regulatory commitments contained in this letter.

If you have any questions concerning, please contact William Drews, Regulatory Assurance Manager, at (315) 349-6562.

Sincerely,

William C. Drews
Regulatory Assurance Manager

WD/mh

Enclosure: SEP-IST-007 Inservice Testing (IST) Program Plan

cc:
NRC Regional Administrator, Region I
NRC Resident Inspector
NRC Project Manager

JAFP-18-0078
Enclosure

SEP-IST-007 Inservice Testing (IST) Program Plan

(189 Pages)

Exelon Nuclear Generation, LLC

200 Exelon Way
Kennett Square, PA 19348

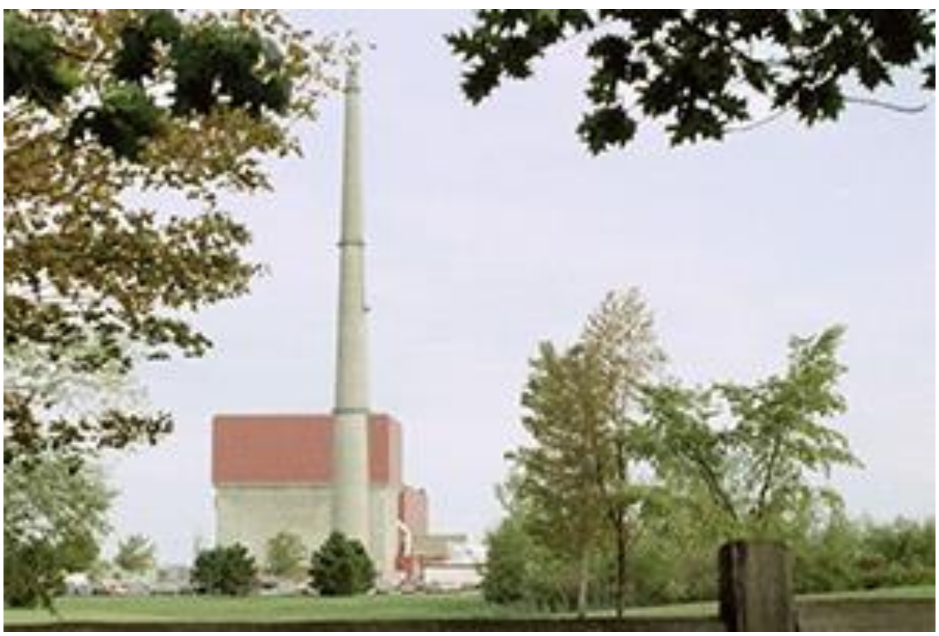
James A. FitzPatrick Nuclear Power Plant
Unit Docket Number **50-333**
PO Box 110
Lycoming, NY 13093
Commercial Service Date: October 17, 1974

SEP-IST-007

Inservice Testing (IST) Program Program Plan

5th Ten-Year Interval

June 1, 2018 – September 30, 2027



Revision 10
June 1, 2018

SEP-IST-007
REVISION RECORD

Effective Date	Revision Description	Sign & Date		
		Prepared: Site IST Engineer	Reviewed: Corporate IST Engineer	Approved; Engr. Programs Manager
06/01/2018	Revision 10- Major rewrite to implement the 5th Ten-Year Interval Update and adopt ER-AA-321 program plan format	[illegible] 5/24/2018	[illegible] 5/30/18	M. Faivus [illegible] 5-31-2018

SEP-IST-007

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 Purpose

The purpose of this Inservice Testing (IST) Program Plan is to provide a summary description of the James A. FitzPatrick Nuclear Power Plant (JAF) IST Program in order to document its compliance with the requirements of 10 CFR 50.55a(f) for the 5th ten-year IST interval.

1.2 Scope

This Inservice Testing Program Plan identifies all of the testing performed on the components included in the JAF Inservice Testing (IST) Program for the 5th ten-year IST interval, which began on June 1st, 2018 and is scheduled to end on September 30, 2027.

The Code of Federal Regulations, 10 CFR 50.55a(f)(4), requires that throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the inservice test requirements set forth in the ASME OM Code and addenda that are incorporated by reference in paragraph 10 CFR 50.55a(b)(3) for the initial and each subsequent 120-month interval.

Based on the start date identified above, the IST Program for the 5th ten-year interval is required by 10 CFR 50.55a(f)(4)(ii) to comply with the requirements of the ASME OM Code-2004, Code for Operation and Maintenance of Nuclear Power Plants, including addenda through the OM-2006*,* except where relief from such requirements has been granted in writing by the NRC.

The scope of the OM Code is defined in paragraph ISTA-1100 as applying to:

(a) pumps and valves that are required to perform a specific function in shutting down a reactor to the safe shutdown condition, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident;

(b) pressure relief devices that protect systems or portions of systems that perform on or more of the functions listed in (a), above; and

(c) dynamic restraints (snubbers) used in systems that perform one or more of the functions listed in (a).

NOTE: This IST Program Plan addresses only those components included in (a) and (b) above. Dynamic restraints (snubbers) are addressed in a separate test program.

In order to determine the scope of the IST Program at JAF, an extensive scope evaluation was performed. This scope evaluation determined all of the functions required to be performed by all ASME Class 1, 2 and 3 systems in shutting down the reactor to the safe shutdown condition, in maintaining the safe shutdown condition or in mitigating the consequences of an accident. The determination of those functions was accomplished by a thorough review of licensing bases documents such as the UFSAR/FSAR, Plant Technical Specifications and Technical Specification Bases documents, etc. Next, a component-by-component review was performed to determine what function each pump and valve in the system was required to perform in order to support the safety function(s) of the system or subsystem. The results of these efforts are documented in the Station's IST Bases Document. In addition to a description of each component's safety function(s), the Bases Document identifies the tests and examinations that are performed on each component to provide assurance that they will be operationally ready to perform those safety function(s). The Bases Document identifies those ASME Class 1, 2, and 3 pumps and valves that are in the scope of the IST Program, including those that do and those that do not have required testing. It also identifies those ASME Class 1, 2 and 3 pumps and valves that are outside the scope of the IST Program on the basis that they are not required to perform any specific safety function.

As stated at the beginning of this Section, the scope of this IST Program Plan is to identify all of the testing performed on those components within the scope of the IST Program. This is accomplished primarily by means of the IST Pump and IST Valve Tables contained in Attachments 14 and 15. The remaining Sections and Attachments of this document provide support information to that contained in the Tables. Components that do not require testing are not included in the IST Program Plan document.

In addition to those components that are required to perform specific safety function(s), the scope evaluation often determines that there are also ASME Safety Class 1, 2 and 3 components that are not required to perform a licensing-based safety function but which, nonetheless, may be relied upon to operate to perform a function with some significance to safety. It may also identify non-ASME Safety Class pumps or valves that have a safety function or may be relied upon to operate to perform a function with some significance to safety. None of these components are required by 10 CFR 50.55a to be included in the IST Program. However, such components may require testing in a manner which demonstrates their ability to perform their functions commensurate with their importance to safety per the applicable portions of 10 CFR 50, Appendix A or B. One option is to include pumps or valves that fit these conditions in the IST Program as augmented components.

JAF is licensed with Hot Shutdown as the safe shutdown condition. Therefore, the scope of the IST Program must include, as a minimum, all of those ASME Class 1, 2, and 3 pumps and valves which are required to shut down the Reactor to the Hot Shutdown condition, maintain the Hot Shutdown condition, or mitigate the consequences of an accident.

1.3 <u>Discussion</u>

A summary listing of all the pumps and valves that are tested in accordance with the IST Program is provided in the IST Pump and IST Valve Tables contained in Attachments 14 and 15. The Pump and Valve Tables also identify each test that is performed on each component, the frequency at which the test is performed, and any Relief Request or Technical Position applicable to the test. For valves, the Valve Table also identifies any Cold Shutdown Justification or Refueling Outage Justification that is applicable to the required exercise tests. Additional information is provided for both pumps and valves. All of the data fields included in the IST Pump and Valve Tables are listed and described in Sections 2 and 3 of this document.

Following Sections 2 and 3 are several Attachments which provide information referenced in the Pump and Valve Tables.

Attachment 1 includes a listing of System and P&IDs on which a depiction of the pump or valve may be located.

Attachment 2 provides an index of the Pump Relief Requests that apply to any of the pumps in the IST Program for this ten-year interval.

Attachment 3 includes a copy of each of those Relief Requests.

Attachment 4 provides an index of the Valve Relief Requests that apply to any of the valves in the IST Program for this ten-year interval.

Attachment 5 includes a copy of each of those Relief Requests.

Attachment 6 contains the Safety Evaluation Report(s) (SER) that document approval of the Relief Requests contained in Attachments 3 and 5.

Attachment 7 includes a list of the ASME OM Code Cases that are being invoked for this ten-year interval.

Attachment 8 provides an index of Cold Shutdown Justifications that apply to the exercise testing of any valves in the IST Program for this ten-year interval.

Attachment 9 includes a copy of each of those Cold Shutdown Justifications.

Attachment 10 provides an index of Refueling Outage Justifications that apply to the exercise testing of any valves in the IST Program for this ten-year interval.

Attachment 11 includes a copy of each of those Refueling Outage Justifications.

Attachment 12 provides an index of Technical Positions that apply to the IST Program for this ten-year interval. Technical Positions provide detailed information regarding how Exelon satisfies certain ASME OM Code requirements, particularly when the Code requirement may be ambiguous or when multiple options for implementation may be available. Technical Positions do not take exception to or provide alternatives to Code requirements.

Attachment 13 includes a copy of each Technical Position listed in Attachment 12.

As described previously, Attachments 14 and 15 include the IST Pump and Valve Tables.

Attachment 16 provides a listing of Check Valve Condition Monitoring (CVCM) Program Plans. CVCM program plans are maintained in the IST program notebook located on the JAF network.

This IST Program Plan is a quality-related document and is controlled and maintained in accordance with approved Exelon Corporate Engineering and Records Management procedures.

1.4 References

1.4.1 Title 10, Code of Federal Regulations, Part 50, Section 55a (10 CFR 50.55a)

1.4.2 ASME OM Code-2004, Code for Operation and Maintenance of Nuclear Power Plant Components, including Addenda through OM 2006.

1.4.3 James A. FitzPatrick Technical Specification, 5.5.7

1.4.4 Exelon Corporation Administrative Procedure ER-AA-321, Administrative Requirements for Inservice Testing

2.0 INSERVICE TESTING PLAN FOR PUMPS

2.1 Pump Inservice Testing Plan

The James A. FitzPatrick Inservice Testing Program for Pumps meets the requirements of Subsections ISTA and ISTB of the ASME OM Code-2004 with OMb 2006 addenda, with the exception of those specific applications identified in the Relief Requests contained in Attachment 3.

2.2 IST Plan Pump Table Description

The pumps included in the James A. FitzPatrick Inservice Testing Program are listed in Attachment 14. The information contained in that table identifies those pumps required to be tested to the requirements of the ASME OM Code, the parameters measured, associated Relief Requests and comments, and other applicable information. The column headings for the Pump Table are listed below with an explanation of the content of each column.

Pump ID	The unique identification number for the pump, as designated on the System P&ID or Flow Diagram
Description	The descriptive name for the pump
Class	The ASME Safety Class (i.e., 1, 2 or 3) of the pump. Non-ASME Safety Class pumps are designated "N/A".
Group	A or B, as defined in Reference 1.4.2.
DWG No.	The Piping and Instrumentation Diagram or Flow Drawing on which the pump is shown
CO-ORD	The coordinates on the P&ID or Flow Diagram where the pump is shown.
Pump Type	An abbreviation used to designate the type of pump: C Centrifugal PDN Positive Displacement - Non-Reciprocating PDR Positive Displacement - Reciprocating VLS Vertical Line Shaft
Driver	The type of driver with which the pump is equipped. A Air-motor D Diesel M Motor (electric) T Turbine (steam)

<u>Test</u>

Lists each of the test parameters which are required to be measured for the specific pump. These include:

N	Speed (for variable speed pumps, only)
ΔP	Differential Pressure
P	Discharge Pressure (positive displacement pumps)
Q	Flow Rate
V_d	Vibration (displacement)
V_v	Vibration (velocity)

<u>Test Frequency</u>

An abbreviation which designates the frequency at which the associated test is performed:

Q	Quarterly
Y2	Once every 2 years

NOTE: All tests are performed at the frequencies specified by Code unless specifically documented by a Relief Request.

<u>Relief Request</u>

Identifies the number of the Relief Request applicable to the specified test.

3.0 INSERVICE TESTING PLAN FOR VALVES

3.1 Valve Inservice Testing Plan

The James A. FitzPatrick Inservice Testing Program for Valves meets the requirements of Subsections ISTA and ISTC of the ASME OM Code-2004 with OMb-2006 addenda, with the exception of those specific applications identified in the Relief Requests contained in Attachment 5.

3.2 IST Plan Valve Table Description

The valves included in the James A. FitzPatrick Inservice Testing Program are listed in Attachment 15. The information contained in that table identifies those valves required to be tested to the requirements of the ASME OM Code, the testing methods and frequency of testing, associated Relief Requests, comments, and other applicable information. The column headings for the Valve Table are delineated below with an explanation of the content of each column.

Valve ID — The unique identification number for the valve, as designated on the System P&ID or Flow Diagram.

Description — The descriptive name for the valve.

Vlv Type — An abbreviation used to designate the body style of the valve:

3W	3-Way
4W	4-Way
BAL	Ball
BTF	Butterfly
CK	Check
DIA	Diaphragm
GA	Gate
GL	Globe
PLG	Plug
RPD	Rupture Disk
RV	Relief
SCK	Stop-Check
SHR	Shear (SQUIB)
XFC	Excess Flow Check

Actu Type	An abbreviation which designates the type of actuator on the valve. Abbreviations used are: AO Air Operator DF Dual Function (Self and Power) EXP Explosive HO Hydraulic Operator M Manual MO Motor Operator SA Self-Actuating SO Solenoid Operator
P&ID	The Piping and Instrumentation Diagram or Flow Drawing on which the valve is shown.
Coord	The coordinates on the P&ID or Flow Diagram where the valve is shown.
Class	The ASME Safety Class (i.e., 1, 2 or 3) of the valve. Non-ASME Safety Class valves are designated by “N/A”.
Positions Norm/Safe	Abbreviations used to identify the normal and safety-related positions for the valve. Abbreviations used are: AI As Is C Closed CKL Closed/Actuator Key Locked D De-energized D/E De-energized or Energized E Energized LC Locked Closed LO Locked Open LT Locked Throttled O Open O/C Open or Closed OKL Open/Actuator Key Locked SYS System Condition Dependent T Throttled
Cat	The ASME Code category or categories of the valve as defined in Reference 1.4.2.
Active/Passive	“A” or “P”, used to designate whether the valve is active or passive in fulfillment of its safety function. The terms “active valves” and “passive valves” are defined in Reference 1.4.2.

<u>Testing Requirements</u>

- <u>Test</u>

A listing of abbreviations used to designate the types of testing which are required to be performed on the valve based on its category and functional requirements. Abbreviations used are:

BDC	Bidirectional Check Valve test (non-safety related closure test)
BDO	Bidirectional Check Valve test (non-safety related open test)
CC[2]	Check Valve Exercise Test - Closed
CO[2]	Check Valve Exercise Test - Open
CP[2]	Check Valve Partial Exercise Test
DT	Category D Test
EC	Exercise Test – Closed (manual valve)
EO	Exercise Test – Open (manual valve)
FC	Fail-Safe Exercise Test - Closed
FO	Fail-Safe Exercise Test - Open
LT[1]	Leak Rate Test
PI	Position Indication Verification Test
RT	Relief Valve Test
SC	Exercise Closed (without stroke-timing)
SO	Exercise Open (without stroke-timing)
SP	Partial Exercise (Cat. A or B)
STC	Exercise/Stroke-Time Closed
STO	Exercise/Stroke-Time Open

[1] A third letter, following the “LT” designation for leakage rate test, may be used to differentiate between the tests. For example, Appendix J leak tests will be designated as “LTJ”, low pressure (non-Appendix J) leak tests as “LTL”, and high pressure leak tests as “LTH”.

[2] Three letter designations should be used for check valve tests to differentiate between the various methods of exercising check valves. The letter following “CC”, “CO” or “CP” should be “A” for acoustics, “D” for disassembly and inspection, “F” for flow indication, “M” for magnetics, “R” for radiography, “U” for ultrasonics, or “X” for manual exercise.

- Freq — An abbreviation which designates the frequency at which the associated test is performed. Abbreviations used are:

  AJ Per Appendix J
  CM Per Check Valve Condition Monitoring Program
  CS Cold Shutdown
  M[*n*] Once Every *n* Months
  Q Quarterly
  RR Refuel Outage
  R[*n*] Once Every *n* Refuel Outages
  SA Sample Disassemble & Inspect
  TS Per Technical Specification
  Requirements
  Y[*n*] Once Every *n* Years

- RR — Identifies the number of the Relief Request applicable to the specified test.

- Justification — A cross-reference to the applicable Cold Shutdown Justification or Refuel Outage Justification which describes the reasons why reduced-frequency exercise testing is necessary for the applicable valve.

Comments — Any appropriate reference or explanatory information (e.g., technical positions, etc.).

SECTION 4.0

ATTACHMENTS

ATTACHMENT 1

SYSTEM AND P&ID LISTING

SYSTEM #	SYSTEM NAME	DRAWING #
01-125	Standby Gas Treatment	FM-48A
02	Automatic Depressurization	FM-29A
02-2	Reactor Water Recirculation	FM-26A
02-3	Nuclear Boiler Instrumentation	FM-47A
03	Control Rod Drive	FM-27B
07	Neutron Tip Monitors	FM-119A
10	Residual Heat Removal	FM-20A,B
11	Standby Liquid Control	FM-21A
12	Reactor Water Cleanup	FM-24A
13	Reactor Core Isolation Cooling	FM-22A
14	Core Spray	FM-23A
15	Reactor Building Closed Loop Cooling	FM-15A,B
16-1	Leak Rate Analyzer	FM-49A
20	Radioactive Waste	FM-17A
23	High Pressure Cooling Injection	FM-25A
27	Containment Atmosphere Dilution	FM-18A,B,D
29	Main Steam	FM-29A
34	Feedwater	FM-34A
39	Breathing, Instrument & Service Air	FM-39A
46	Service & Emergency Service Water	FM-46A,B
66	Reactor Building Service Ventilation (Service Water)	FB-10H
70	Control Room Service & Chilled Water	FB-35E

ATTACHMENT 2

PUMP RELIEF REQUEST INDEX

RELIEF REQUEST NUMBER	**RELIEF REQUEST TITLE**	**APPROVAL DATE**
PRR-001	Core Spray Pump Suction Pressure Gauge Range	4/13/2018
PRR-002	RHRSW Smooth Running Pumps	4/13/2018
PRR-003	RHR Pump Vibration Alert Range	4/13/2018
PRR-004	Code Case OMN-21	4/13/2018

ATTACHMENT 3

PUMP RELIEF REQUESTS

PRR-01 CORE SPRAY PUMP SUCTION PRESSURE GAUGE RANGE

PLANT/UNIT:	James A. Fitzpatrick (JAF) Nuclear Power Plant
INTERVAL:	5th Interval beginning June 1, 2018, and ending September 30, 2027
COMPONENTS AFFECTED:	14P-1A Core Spray Pump 14P-1B Core Spray Pump
CODE EDITION AND ADDENDA:	ASME OM Code-2004 Edition with Addenda through OMb-2006

REQUIREMENTS: ISTB-3500, Data Collection, paragraph ISTB-3510, General, (a) *Accuracy*, states, in part: “Instrument accuracy shall be within the limits of Table ISTB-3510-1…”

ISTB-3500, Data Collection, paragraph ISTB-3510, General, (b) *Range*, (1), states: “The full-scale range of each analog instrument shall be not greater than three times the reference value.”

ISTB-3500, Data Collection, Table ISTB 3510-1, Required Instrument Accuracy, requires pressure and differential pressure accuracy to be ± 2% for Group A and Group B tests.

REASON FOR REQUEST: Pursuant to 10 CFR 50.55a(z)(2), an alternative is requested from the ASME OM Code paragraph(s) ISTB 3500, Table ISTB 3510-1, that requires the suction and differential pressure accuracy to be within ± 2% of full scale range and ISTB 3510(b)(1) that requires the full-scale range of an analog instrument to be less than three times the reference value. The basis of this request is that the Code requirements present an undue hardship without a compensating increase in the level of quality or safety.

PROPOSED ALTERNATIVE AND BASIS: JAF is proposing to use the existing installed plant suction pressure gauges to determine the pump differential pressure for testing the Core Spray pumps for the Group B pump tests. The comprehensive tests are performed using Measuring and Test Equipment (M&TE).

The differential pressure for the Core Spray pumps is calculated using the installed suction and discharge pressure gauges. The suction pressure gauge is designed to provide adequate suction pressure indication during all expected operating conditions. The suction pressure gauge full-scale range of 72.28 pounds per square inch gauge (psig), [from 25 inches of mercury (12.28 psig) vacuum to 60 psig], is sufficient for a post-accident condition when the torus is at the maximum accident pressure. This, however, exceeds the range limit of 15 psig for the suction pressure under the test condition of approximately 5 psig.

PRR-01 CORE SPRAY PUMP SUCTION PRESSURE GAUGE RANGE

PROPOSED ALTERNATIVE AND BASIS (Continued)

The installed discharge pressure instrument loop is calibrated to within ± 2% full-scale accuracy. However, the installed suction pressure instrument loop is calibrated to within ± 2.4% full-scale accuracy. The full-scale range of the pump discharge pressure instrumentation loop is 500 psig. Pump discharge pressure during testing is typically 300 psig. Thus, the maximum variation due to inaccuracy in measured suction pressure is ± 1.7 psi and in measured discharge pressure is ± 10 psi. Therefore, the differential pressure (typical [P_{disch} - P_{suct}]) would be (300 – 5), which is 295 ± 11.7 psi for an inaccuracy of 4.0%. If the full-scale range of the suction pressure gauge was within the Code allowable of 3 times the reference value, or 15 psig, the maximum variation due to inaccuracy in measured suction pressure would be ± 0.3 psi and the resulting differential pressure measurement would be 295 ± 10.3 psi for an inaccuracy of 3.5%.

The decrease in inaccuracy of 0.5%, by using the Code compliant gauge, is insignificant and does not warrant the additional manpower and exposure required to change the suction pressure gauge quarterly for test purposes.

In addition, the Code would allow a full-scale range for the discharge pressure measurement of 900 psig for the typical 300 psig discharge pressure. This would translate into a differential pressure measurement of 295 ± 18.3 psig or an inaccuracy of 6.2%, if the installed instrumentation met the Code requirements of 0 – 15 psig for the suction pressure gauge and 0 – 900 psig for the discharge pressure gauge. The existing measurement is significantly better than the maximum Code allowable inaccuracy.

Based on the hardship resulting from implementing the OM Code requirements without a compensating increase in the level of quality and safety, this proposed alternative is requested pursuant to 10 CFR 50.55a(z)(2).

DURATION: 5^{th} IST Interval, beginning June 1, 2018, and ending September 30, 2027

PRECEDENT: This proposed 10 CFR 50.55a alternative (for the instrument range) was previously authorized for use as PRR-02 at JAF during the fourth ten-year IST interval via the NRC's safety evaluation, dated November 27, 2007. (ADAMS Accession No. ML072910422)

PRR-002 RHRSW Smooth Running Pumps

PLANT/UNIT: James A. Fitzpatrick (JAF) Nuclear Power Plant

INTERVAL: 5th Interval beginning June 1, 2018, and ending September 30, 2027

COMPONENTS AFFECTED:
10P-1A, Residual Heat Removal Service Water (RHRSW) Pump
10P-1B, RHRSW Pump
10P-1C, RHRSW Pump
10P-1D, RHRSW Pump

The RHRSW pumps are vertical line shaft pumps that provide cooling water to the Residual Heat Removal (RHR) Heat Exchangers during a design basis event. These smooth running pumps are in the JAF inservice testing (IST) program and are ASME OM Code Group A pumps.

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OMb-2006

REQUIREMENTS: ISTB-3300, Reference Values, states that “Reference values shall be obtained as follows:” ISTB-3300(a), states: “Initial reference values shall be determined from the results of testing meeting the requirements of ISTB-3100, Preservice Testing, or from the results of the first inservice test.”

REASON FOR REQUEST: Pursuant to 10 CFR 50.55a(z)(1), an alternative is requested from the ASME OM Code ISTB requirements referenced above, related to the vibration reference value (V_r), for the pumps that are identified as smooth running pumps. The basis of this request is that the alternative reference values provide an acceptable level of quality or safety.

The smooth running pumps in the JAF Nuclear Power Plant IST Program have at least one vibration reference value (V_r) that is currently less than 0.05 inches per second (ips). A small value for V_r produces a small acceptable range for pump operation. The OM Code Acceptable Range limit for pump vibrations from Table ISTB-5221-1, Vertical Line Shaft Centrifugal Pump Test Acceptance Criteria, for both the Group A test and Comprehensive test is ≤ 2.5 V_r. Based on a small acceptable range, a smooth running pump could be subject to unnecessary corrective action if it exceeds this limit.

ISTB-6200, Corrective Action, (a), Alert Range, states; “If the measured test parameter values fall within the alert range of Table ISTB-5121-1, Table ISTB-5221-1, Table ISTB-5321-1 or Table ISTB-5321-2, as applicable, the frequency of testing specified in ISTB-3400 shall be doubled until the cause of the deviation is determined and the condition is corrected.”

PRR-002 RHRSW Smooth Running Pumps

REASON FOR REQUEST (Continued):	For very small reference values for vibrations, a significant portion of the reading can be from flow variations, hydraulic noise, and instrument error, which can affect the repeatability of subsequent measurements. Also, experience gathered by the JAF Nuclear Power Plant Predictive Maintenance (PdM) group has shown that changes in vibration levels in the range of 0.05 ips do not normally indicate significant degradation in pump performance. In order to avoid unnecessary corrective actions, a minimum value for V_r of 0.05 ips is proposed. This minimum value would be applied to individual vibration locations, for the RHRSW pumps identified in this alternative request, with reference vibration values less than 0.05 ips.
PROPOSED ALTERNATIVE AND BASIS:	In lieu of applying the reference values required by ISTB-3300, for smooth running pumps with a measured reference value below 0.05 ips, JAF will apply a minimum value for V_r of 0.05 ips for the particular vibration measurement location. This minimum value would be applied to individual vibration locations for the RHRSW pumps, identified in this alternative request. The subsequent test results for that location will be compared to an Alert Range limit of 0.125 ips and a Required Action limit of 0.300 ips. These ranges, resulting from the proposed V_r of 0.05 ips and using the existing OM Code multipliers, shall be applied to vibration test results during both Group A and Comprehensive tests. Therefore, the smallest OM Code Alert Range limit for any IST pump vibration location would be > 2.5 times V_r, or 0.125 ips, which is within the "fair" range of the "General Machinery Vibration Severity Chart" provided by IRD Mechanalysis, Inc. Likewise, the smallest OM Code Required Action limit for any IST Pump vibration location for which the pump would be inoperable would be > 6 times V_r, or 0.300 ips. For comparison purposes, ASME Section XI, Table IWP-3100-2, "Allowable Ranges of Test Quantities," specifies a vibration Acceptable Range limit of 1.0 mil for a displacement reference value ≤ 0.5 mils. In velocity units, a displacement reference value of 0.5 mils is equivalent to 0.047 ips for an 1800 rpm pump and 0.094 ips for a 3600 rpm pump. The minimum reference value proposed (0.05 ips) for smooth-running pumps is roughly equal to the ASME Code Section XI IWP reference value for an 1800 rpm pump and more conservative than the reference value for a 3600 rpm pump. In addition to the requirements of OM Code Subsection ISTB for IST, the pumps in the JAF Nuclear Power Plant IST Program are also included in the JAF Nuclear Power Plant PdM Program. The JAF Nuclear Power Plant PdM Program currently employs predictive monitoring techniques such as: vibration monitoring and analysis beyond that required by Subsection ISTB, bearing temperature trending, oil sampling and analysis, and/or thermography and analysis, as applicable.

PRR-002 RHRSW Smooth Running Pumps

PROPOSED ALTERNATIVE AND BASIS (Continued):

If the measured parameters are outside the normal operating range or are determined by analysis to be trending toward an unacceptable degraded state, appropriate actions are taken that may include: initiate a Condition Report (CR), increase monitoring to establish a rate of change, review component specific information to identify the cause, and remove the affected pump from service to perform maintenance.

It should be noted that the pumps in the IST Program will remain in the JAF PdM program even if certain pumps have very low vibration readings and are considered to be smooth running pumps.

Using the provisions of this request as an alternative to the required reference value determination specified in ISTB-3300, the vibration acceptance criteria, based on the alternative reference value, provides an acceptable level of quality and safety since the alternative provides reasonable assurance of pump operational readiness and the ability to detect pump degradation. Therefore, pursuant to 10 CFR 50.55a(z)(1), JAF requests use of the proposed alternative to the specific ISTB Code requirements identified in this 10CFR50.55a Request.

DURATION:

5th IST Interval, beginning June 1, 2018, and ending September 30, 2027

PRECEDENCE:

1. This 10CFR 50.55a request was previously approved for the Interval 4 IST Program at JAF via relief request PRR-04 in NRC SER dated November 27, 2007 (ADAMS Accession No. ML072910422).

2. Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Evaluation of Inservice Testing (IST) Pump Relief Request No. 8 Revisions 1K and 2I, respectively, NRC SER dated December 27, 2004 (ADAMS Accession No. ML043430042).

REFERENCES:

1. General Machinery Vibration Severity Chart provided by IRD Mechanalysis, Inc.

2. ASME Section XI, Subsection IWP, Inservice Testing of Pumps in Nuclear Power Plants, (IWP-3200-IWP-3400),1986 Edition

PRR-003 RHR PUMP VIBRATION ALERT RANGE

PLANT/UNIT:	James A. Fitzpatrick (JAF) Nuclear Power Plant
INTERVAL:	5^{th} Interval beginning June 1, 2018, and ending September 30, 2027
COMPONENTS AFFECTED:	10P-3A, Residual Heat Removal (RHR) Pump 10P-3B, RHR Pump 10P-3C, RHR Pump 10P-3D, RHR Pump The four RHR pumps are classified as ASME Class 2 and OM Code Group A pumps.
CODE EDITION AND ADDENDA:	ASME OM Code-2004 Edition with Addenda through OMb-2006
REQUIREMENTS:	ISTB Table ISTB-5121-1, “Centrifugal Pump Test Acceptance Criteria” provides the vibration “Alert Range” low-end absolute limit of 0.325 inches per second (ips) for the Group A and Comprehensive tests.
REASON FOR REQUEST:	Pursuant to 10 CFR 50.55a(z)(2), an alternative is requested from the ASME OM Code, Subsection ISTB, Table ISTB-5121-1 vibration “Alert Range” low-end absolute limit of 0.325 ips requirement during the Group A or biennial comprehensive pump test. The basis of this request is that the Code requirements present an undue hardship without a compensating increase in the level of quality or safety. The increased periodicity of testing, resulting from the 0.325 ips Code requirement is an additional burden to the Operations staff, plant scheduling, and adds unnecessary run time to all RHR pumps. This request is based on analysis of vibration and pump differential pressure data indicating that no pump degradation is taking place.
PROPOSED ALTERNATIVE AND BASIS:	JAF is proposing to use an alternative vibration “Alert Range” low-end absolute limit of 0.408 ips. This provides an alternative method that continues to meet the intended function of monitoring the pump for degradation over time while keeping the required action level unchanged. <u>Pump Testing Methodology</u> The RHR pumps at JAF are tested using a full flow recirculation test line back to the suppression pool for each pump surveillance test (including quarterly group A tests and biennial comprehensive pump tests). These pumps have a minimum flow line (per division) which is used only to protect the pump from overheating when pumping against a closed discharge valve. The minimum flow line isolation valve for each division is initially open when the pump is started, and flow is recirculated through

PRR-003 RHR PUMP VIBRATION ALERT RANGE

PROPOSED ALTERNATIVE AND BASIS (Continued):

the minimum flow line back to the suppression pool. Then, the full flow test line isolation valve is throttled open to establish flow through the full flow recirculation test line. The minimum flow line is then automatically isolated and all flow is directed through the full flow test line for the IST test. Based on the full flow test line configuration, this test methodology results in flow-induced, broadband vibration readings greater than 0.325 ips, but less than the required action limits.

The RHR system is operated in the same manner and under the same conditions for each IST test, regardless of whether JAF is operating or shutdown. Consequently, the pumps will experience the same potential for flow-induced, broadband vibration whenever they are tested, whether JAF is operating or shutdown. As a result, this alternative is proposed for the inservice testing of RHR pumps when vibration measurements are required.

<u>NRC Staff Document NUREG/CP-0152</u>

NRC Staff document NUREG/CP-0152, entitled “Proceedings of the Fourth NRC/ASME Symposium on Valve and Pump Testing,” dated July 15-18, 1996, included a paper prepared by the NRC staff, entitled Nuclear Power Plant Safety Related Pump Issues. That paper presented four key components that should be addressed in an alternative request of this type to streamline the review process. These four key components are as follows:

I. The licensee should have sufficient vibration history from the inservice testing which verifies that the pump has operated at the vibration level for a significant amount of time, with any “spikes” in the data justified.
II. The licensee should have consulted with the pump manufacturer or vibration expert about the level of vibration the pump is experiencing to determine if the pump operation is acceptable.
III. The licensee should describe attempts to lower the vibration below the defined code absolute levels through modifications to the pump.
IV. The licensee should perform a spectral analysis of the pump driver system to identify all contributors to the vibration levels.

The following is a discussion of how these four key components are addressed for this alternative request.

I. <u>Vibration History</u>

a. <u>Testing Methods and Code Requirements</u>

Elevated vibration levels on the RHR pumps has been a condition that has existed since original installation. Prior to 1998, testing was measured in displacement (mils). These

PRR-003 RHR PUMP VIBRATION ALERT RANGE

PROPOSED ALTERNATIVE AND BASIS (Continued):

readings were taken in two directions, horizontal in-line with pump flow and horizontal perpendicular to pump flow. In 1998, JAF entered the third 10-year interval and implemented ASME/ANSI OM-1989, OM-6, for pump testing. During this interval, IST vibration data was taken in displacement (mils). Additional data was also taken in velocity (ips), with an additional data point in the axial direction. At that time, the velocity data points were used as information only. Upon the fourth 10 year interval at JAF, ASME OM Code 2001 Edition with 2003 Addenda was adopted. With this adoption, it was determined that vibration measurements in velocity would be a much better indicator of pump condition and would be beneficial in terms of early identification of degradation. Therefore, data exists for two vibration points on each RHR pump from January 1986 to August 1998 in mils. Data from May of 1999 to present is in the form of ips at three vibration data points. Various analyses of this data are included as Figures 1 through10 within this alternative request.

b. Review of Vibration History Data

IST trends of RHR Pump vibration (Figures 1 - 4), which include data from 1999 through present, show some readings to be at or above current IST vibration alert criteria. From Figures 9 and 10, it can be seen that, 10P-3B and 10P-3C have exceeded the current OM Code Alert criteria of 0.325 ips.

RHR pump differential pressure trends (Figures 5 - 8) illustrate the differential pressure data during the same time period as the vibration Figures 1 - 4. These graphs show a step change in flow around the 2009 time frame. This change is due to surveillance test changes in which test flow was lowered. The change was made after engineering analysis resulted in revised pump flow criterion. These trends do not show any signs of hydraulic degradation.

A review of the maintenance history for all four pumps and motors shows very minimal maintenance has been performed beyond the normally scheduled preventive maintenance. The only maintenance deemed to have the potential to affect vibration values is that of mechanical seal replacement. RHR pumps 10P-3A and 10P-3C both had mechanical seals replaced in 2009. Post work testing of these replacements did not show any change in vibration values.

PRR-003 RHR PUMP VIBRATION ALERT RANGE

PROPOSED ALTERNATIVE AND BASIS (Continued):

Average run times for each RHR pump per cycle is approximately 200-300 hours. Run times of this nature, combined with the pumps and motors being built and maintained to the nuclear quality standards, are considered to have a low likelihood for significant wear related degradation.

c. Review of “Spikes” in Vibration Data

Trends of recent vibration history (4th Interval) do not show any significant spikes above baseline levels. Instead, all values are seen as fairly consistent and there have been no significant degrading trends associated with vibration data for the past 16 years.

While the overall vibration trends have been steady, when compared to the OM Code, recent vibration data points have exceeded the OM Code “Alert” limit of 0.325 ips. RHR pump 10P-3B exceeded this value once in 2010 and once in 2014. Also, RHR pump 10P-3C exceeded this value once in 2012 and twice in 2016. Without the relief, for the 4th IST Interval at JAF, granted by NRC in the Safety Evaluation dated September 15, 2014, these pumps would be on increased frequency testing even though there has been no pump degradation.

II. Consultation – Pump Manufacturer/Vibration Expert

During the initial investigation for the cause of the failed vibration acceptance criteria, Mancini Consulting Services (an industry pump expert) and Flowserve (the pump vendor) were consulted for input.

Each RHR pump motor is vertically mounted to the pump casing, with the piping entering and exiting the pump casing horizontally. The RHR pump motors weigh approximately 6500 pounds and operate at 1800 revolutions per minute (RPM). Other motor specifications include: 1000 Horsepower (Hp), 3 Phase, 60 Hertz (Hz), 4000 Volts. The pump casing, weighing 6100 pounds, is mounted on a reinforced floor pad.

The 20-inch suction piping enters the room level with the pump centerline. An additional 20-inch line tee’s into the suction piping approximately 5 feet from the pump. The 16-inch discharge piping leaves the pump on the same plane as the suction piping but then elbows 90 degrees vertically 6 feet from the pump. This is then followed by a discharge check valve and an isolation valve on the vertical run. See figures 11 and 12 for the isometric layout of the suction and discharge piping near the RHR pumps.

PRR-003 RHR PUMP VIBRATION ALERT RANGE

PROPOSED ALTERNATIVE AND BASIS (Continued):

Vibration monitoring points are located on the RHR motor/pump assembly. Points V1, V2, and V3 (Vaxial) are the specified locations for IST data collection. V1 is taken in the vertical direction in line with pump flow. V2 is taken 90 degrees from V1 perpendicular to pump flow. V3 is taken 90 degrees from the V1 and V2 plane, on the underside of the motor.

Resonance testing was performed on all four RHR pump housings in the V1, V2, and V3 directions. Analysis has shown that the contributing cause of vibration in the V1 and V2 directions is from broadband peaks in the spectrum between 85.1 to 102.7 Hertz. These frequencies fall in the area of pump operation as these pumps are of a 3-vane design. In the case of the RHR pumps, vane pass frequency excitations are influencing vibration measurements.

III. Attempts to Lower Vibration

Prior to completely understanding the cause of the vibration, it was thought that after the adjustment of testing flow in early 2009, vibrations would decrease. Through the recent analysis of the vibration spectrum, the structural resonance and the running speed peaks were confirmed. This analysis indicated that the running speed spectral peaks are consistent over years of testing. With the resonant frequency being considered as a significant contributor to exceeding the alert vibration range, options to reduce resonance reside in stiffening the pump or an internal design change (i.e. modifying to a 5-vane design).

JAF initially pursued a path to add additional stiffening to the pump and piping system. With the addition of supports, a new seismic analysis would be required for each RHR pump and the associated piping. Due to the complexity and resources needed for a new analysis, combined with the industry OE (reference Fermi 2 approved submittal ML101670372) which shows that stiffening operations also lend to the potential for vibrations to be transferred to the surrounding piping, efforts were ceased.

Major modifications, such as to add stiffening to the pump/motor system or changing the pump to a 5-vane design, when the pumps are not seen as degrading, are not deemed to result in an increase in the level of safety.

IV. Spectral Analysis

ASME OM Code 2004 Edition/2006 Addenda paragraph ISTB-6400 states: "If the reference value of a particular parameter being measured or determined can be significantly influenced by other related conditions, then these conditions shall be analyzed[1] and documented in the record of tests. (See ISTB-9000)."

PRR-003 RHR PUMP VIBRATION ALERT RANGE

PROPOSED ALTERNATIVE AND BASIS (Continued):

The footnote (1) for "analyzed" states: "Vibration measurements of pumps may be foundation, driver, and piping dependent. Therefore, if initial readings are high and have no obvious relationship to the pump, then vibration measurements should be taken at the driver, at the foundation, and on the piping and analyzed to ensure that the reference vibration measurements are representative of the pump and the measured vibration levels will not prevent the pump from fulfilling its function."

Spectral analysis was performed showing the total vibration broken down into individual frequencies over a span from 0-1500 Hz. The analysis, which included a historical compilation of the data points that coincide with the IST surveillance testing, shows the broadband vibration is seen occurring at three times the pump running speed. These vibrations may exceed the Code alert criteria, which would trigger the corrective action process and the need to increase the testing frequency.

Spectral data indicates that the overall vibration levels are primarily made up of a spectrum from 85 to 103 Hz due to the vane pass frequency induced by a 3-vane pump at 1800 RPM. As this vibration stems from the design of the pump, all four RHR pumps are susceptible. This vibration is accentuated on 10P-3B and 10P-3C due to the similar piping configurations. Spectral data do not indicate any degradation to the bearings, pump, or motor that would lead to imbalance or misalignment.

Basis for the OM Code Alternative Alert Values

By this alternative request, JAF is proposing to increase the "Alert Range" low-end absolute limit for vibration from 0.325 ips to 0.408 ips for all four RHR pumps in the V1 and V2 directions. The vane pass induced broadband vibration occasionally causes the overall vibration value to exceed 0.325 ips, resulting in the pumps being placed on an increased testing frequency. A new alert acceptance criterion of 0.408 ips coincides with the "warning" level that is already developed per the Predictive Maintenance (PdM) Program. The basis for the 0.408 ips "warning" level came from the Technical Associates of Charlotte recommendations for vertical pumps. The set points recommended were 0.350 ips and 0.525 ips. The PdM program uses those set points as high and low criteria but the program also has two additional levels. These two additional levels split the difference between the suggested set points, resulting in 0.408 ips and 0.466 ips.

PRR-003 RHR PUMP VIBRATION ALERT RANGE

PROPOSED ALTERNATIVE AND BASIS (Continued):

The RHR Pump vendor (Byron Jackson, now owned by Flowserve) did not recommend a specific value regarding the increased vibration alert limit, but stated that the pumps should not be adversely impacted provided that no upward trend existed in the vibration measurement data.

Expert analyses and maintenance reviews have shown that this vibration has not resulted in degradation to the pump or motor. Data trends show that overall vibrations have remained steady since 1998.

The new Alert criterion value allows an alternative measure that still meets the intended function of monitoring the pump for degradation, while leaving the action levels as mandated by the OM Code. The proposed criterion encompasses the previous values that exceeded the Alert level, which would eliminate the unnecessary actions associated with exceeding the OM Code Alert limit when the pump is not seen as degrading. Any corrective actions triggered by vibrations between 0.408 ips to 0.7 ips will result in the same OM Code actions as previously required when exceeding the Alert limit of 0.325 ips.

The vibration specialist at JAF routinely performs a spectral analysis on all data recorded during RHR pump inservice testing. This analysis is in addition to IST total vibration values. The analysis provides additional confidence on the ability to detect degradation at an early stage.

Each RHR pump motor is also monitored through the Preventive Maintenance (PM) and PdM programs. While these actions are intended to prevent degradation, any off-normal or unexpected conditions, will act as an indicator of the early stages of degradation. PM and PdM activities include: annual non-intrusive thermography, annual motor bearing sample, and 10-year internal visual inspection.

<u>Conclusion</u>

Based on the hardship resulting from implementing the OM Code requirements without a compensating increase in the level of quality and safety, this proposed alternative is requested pursuant to 10 CFR 50.55a(z)(2). JAF considers the proposal to use an alternative vibration “Alert Range” low-end absolute limit of greater than or equal to 0.408 ips, while keeping the required action level unchanged, will provide an alternative method that continues to meet the intended function of monitoring the RHR pumps for degradation over time.

DURATION: 5th IST Interval, beginning June 1, 2018, and ending September 30, 2027

PRR-003 RHR PUMP VIBRATION ALERT RANGE

PRECEDENCE:

1. This 10CFR50.55a request was previously approved for the Interval 4 IST Program at JAF via Relief Request PRR-05 in the Safety Evaluation, dated September 15, 2014 (ML14241A329)
2. "Cooper Nuclear Station – RE: Request for Relief (Relief Request RP-06, Revision 2) from the Requirements of ASME Code concerning Inservice Testing of Core Spray Pump CS-P-B," dated February 25, 2004 (ML040560318)
3. "Evaluation of Relief Request PR-2 Associated with Second 10-Year Interval Inservice Testing Program for Pumps and Valves for Byron Station, Units 1 and 2," dated February 19, 2002 (ML020070381)
4. "Fermi 2 – Evaluation of Relief Request Nos: PRR-004, PRR-005, PRR-007, and PRR-010 for the Third 10-Year Interval Inservice Program," dated July 6, 2010 (ML101670372)

PRR-004 CODE CASE OMN-21

PLANT/UNIT:	James A. Fitzpatrick (JAF) Nuclear Power Plant
INTERVAL:	5th Interval beginning June 1, 2018, and ending September 30, 2027
COMPONENTS AFFECTED:	Refer to Table PRR-04
CODE EDITION AND ADDENDA:	ASME OM Code-2004 Edition with Addenda through OMb-2006

REQUIREMENTS: ISTB-5121, "Group A Test Procedure," paragraph ISTB-5121(b) states, in part, that “The resistance of the system shall be varied until the flow rate equals the reference point. … Alternatively, the flow rate shall be varied until the differential pressure equals the reference point…”ISTB-5122, "Group B Test Procedure," paragraph ISTB-5122(c) states, “System resistance may be varied as necessary to achieve the reference point.”

ISTB-5123, "Comprehensive Test Procedure," paragraph ISTB-5123(b) states, in part, that “…the resistance of the system shall be varied until the flow rate equals the reference point. … Alternatively, the flow rate shall be varied until the differential pressure equals the reference point…”

ISTB-5221, "Group A Test Procedure," paragraph ISTB-5221(b) states, in part, that “The resistance of the system shall be varied until the flow rate equals the reference point. … Alternatively, the flow rate shall be varied until the differential pressure equals the reference point…”

ISTB-5222, "Group B Test Procedure," paragraph ISTB-5222(c) states, “System resistance may be varied as necessary to achieve the reference point.”

ISTB-5223, "Comprehensive Test Procedure," paragraph ISTB-5123(b) states, in part, that “The resistance of the system shall be varied until the flow rate equals the reference point. ...Alternatively, the flow rate shall be varied until the differential pressure equals the reference point…”

REASON FOR REQUEST: Pursuant to 10 CFR 50.55a(z)(1), an alternative is proposed to the pump testing reference value requirements of the ASME OM Code. The basis of the request is that the proposed alternative would provide an acceptable level of quality and safety. Specifically, this alternative is requested for all inservice testing of IST Program pumps as listed in attached Table PRR-04, Pumps Affected by Alternative Request PRR-04.

For pump testing, there is difficulty adjusting system throttle valves with sufficient precision to achieve exact flow reference values during subsequent IST tests. Subsection ISTB of the ASME OM Code does not allow for variance from a fixed reference value for pump testing. However, NUREG-1482, “Guidelines for Inservice Testing at Nuclear Power Plants: Inservice Testing of Pumps and Valves and Inservice Examination and

PRR-004 CODE CASE OMN-21

REASON FOR REQUEST (Continued):

Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants," Revision 2, Section 5.3, acknowledges that certain pump system designs do not allow for the licensee to set the flow at an exact value because of limitations in the instruments and controls for maintaining steady flow.

ASME OM Code Case OMN-21, "Alternative Requirements for Adjusting Hydraulic Parameters to Specified Reference Points," provides guidance for adjusting reference flow or differential pressure (ΔP) to within a specified tolerance during inservice testing.

The OM Code Case OMN-21 states, "It is the opinion of the Committee that when it is impractical to operate a pump at a specified reference point and adjust the resistance of the system to a specified reference point for either flow rate, differential pressure or discharge pressure, the pump may be operated as close as practical to the specified reference point with the following requirements. The Owner shall adjust the system resistance to as close as practical to the specified reference point where the variance from the reference point does not exceed + 2% or - 1% of the reference point when the reference point is flow rate, or + 1% or – 2% of the reference point when the reference point is differential pressure or discharge pressure."

The NRC also discusses this ASME Code change in NUREG-1482, Revision 2, Section 5.3.

PROPOSED ALTERNATIVE AND BASIS:

JAF seeks to perform future inservice pump testing in a manner consistent with the requirements as stated in ASME OM Code Case OMN-21. Specifically, testing of all pumps identified in Table PRR-04 will be performed such that the flow rate is adjusted as close as practical to the reference value and within proceduralized limits of +2% / -1% of the reference flow rate or alternatively the differential pressure or discharge pressure is adjusted as close as practical to the reference value and within proceduralized limits of +1% / -2% of the reference pressure or differential pressure.

JAF plant operators will continue to strive to achieve the exact test reference values (flow or differential pressure) during testing. Typical test guidance will be to adjust the reference parameter to the specific reference value with additional guidance that if the reference value cannot be achieved with reasonable effort the test will be considered valid if the steady state flow rate is within the proceduralized limits of +2% / -1% of the reference value or the pressure or differential pressure is within the proceduralized limits of +1% / -2% of the reference value.

The ASME Operation and Maintenance Standards Committee approved Code Case OMN-21 on April 20, 2012, with the NRC representative voting in the affirmative. The applicability of Code Case OMN-21 is the ASME OM Code 1995 Edition through the 2011 Addenda. The language from Code Case OMN-21 has been included in the ASME OM Code, 2012 Edition.

Using the provisions of this request as an alternative to the specific

PRR-004 CODE CASE OMN-21

PROPOSED ALTERNATIVE AND BASIS: (Continued)

requirements of ISTB-5121, ISTB-5122, ISTB-5123, ISTB-5221, ISTB-5222 and ISTB-5223, as described above, will provide adequate indication of pump performance and continue to provide an acceptable level of quality and safety.

Based on the determination that the use of controlled reference value ranges provides an acceptable level of quality and safety, this proposed alternative is requested pursuant to 10 CFR 50.55a(z)(1).

DURATION: 5th Interval beginning June 1, 2018, and ending September 30, 2027

PRECEDENTS: Callaway Plant, Unit 1 – Request for Relief PR-06, Alternative to ASME OM Code Requirements for IST for the Fourth Program Interval – Safety Evaluation dated July 15, 2014 (ML14178A769)

Wolf Creek Generating Station – Request for Relief 4PR-01 for the Fourth 10-Year Inservice Testing Program Interval – Safety Evaluation dated May 15, 2015 (ML15134A002)

REFERENCES: ASME Code Case OMN-21, “Alternate Requirements for Adjusting Hydraulic Parameters to Specified Reference Points”

NUREG-1482, Revision 2, Guidelines for Inservice Testing at Nuclear Power Plants: Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants, Section 5.3, “Allowable Variance from Reference Points and Fixed-Resistance Systems,” dated October 2013 (ML13295A020)

PRR-004 CODE CASE OMN-21

Table PRR-04, Pumps Affected by Alternative Request PRR-04

Pump ID (Units 1 & 2)	Description	Pump Type	Code Class	OM Code Category
11P-2A 11P-2B	Standby Liquid Control Pumps	Positive Displacement	2	Group B
14P-1A 14P-1B	Core Spray Pumps	Centrifugal	2	Group B
10P-3A 10P-3B 10P-3C 10P-3D	Residual Heat Removal (RHR)\Low Pressure Coolant Injection (LPCI) Pumps	Centrifugal	2	Group A
23P-1M	High Pressure Coolant Injection (HPCI) Main Pump	Centrifugal	2	Group B
23P-1B	High Pressure Coolant Injection (HPCI) Booster Pump	Centrifugal	2	Group B
10P-1A 10P-1B 10P-1C 10P-1D	RHR Service Water Pumps	Vertical Line Shaft Centrifugal	3	Group A
46P-2A 46P-2B	Emergency Service Water Pumps	Vertical Line Shaft Centrifugal	3	Group B

ATTACHMENT 4

VALVE RELIEF REQUEST INDEX

RELIEF REQUEST NUMBER	**RELIEF REQUEST TITLE**	**APPROVAL DATE**
VRR-001	TIP CIV STROKE TIME	4/13/2018
VRR-002	ERCV SAMPLING	4/13/2018
VRR-003	CODE CASE OMN-17	4/13/2018
VRR-004	LEAK TEST AT APP J FREQUENCY	4/13/2018

ATTACHMENT 5

VALVE RELIEF REQUESTS

VRR-001 TIP CIV STROKE TIME

PLANT/UNIT: James A. Fitzpatrick (JAF) Nuclear Power Plant

INTERVAL: 5th Interval beginning June 1, 2018, and ending September 30, 2027

COMPONENTS AFFECTED: Traversing In-Core Probe (TIP) Containment Isolation Valves, 07SOV-104A, B, and C

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OMb-2006

REQUIREMENTS: ISTC-5151, Valve Stroke Testing

ISTC-5151(a), states, “Active valves shall have their stroke times measured when exercised in accordance with ISTC-3500.”

ISTC-5151(c), states, “Stroke time shall be measured to at least the nearest second.”

REASON FOR REQUEST: Pursuant to 10 CFR 50.55a(z)(2), an alternative is requested to the stroke time requirements of the ASME OM Code. The basis of this request is that the Code requirements present an undue hardship without a compensating increase in the level of quality or safety.

The Category A, containment isolation solenoid operated valves identified in this request have no safety function in the open direction as they open to allow the passage of the TIP assembly and drive cable for flux mapping operations. These valves have an active safety function in the closed direction in response to a primary containment isolation system signal to seal the TIP guide tubes. Therefore, an exercise test and subsequent stroke time test are only required in the closed direction.

However, the computer control system for the TIP system includes a provision for measuring valve cycle time (opened and closed) and not closure time alone. The sequence opens the subject valve (stroke < 2 seconds), maintains it energized for 10 seconds (including the opening stroke), and de-energizes the valve solenoid allowing the valve to stroke closed (< 2 seconds). The total elapsed time is specified to be $\leq$ 12 seconds.

The design of the TIP control system does not allow for measurement of the closure stroke times of valves 07SOV-104A, B, and C. Measuring the closure stroke times in accordance with the Code would be a hardship since it would require a costly computer control system modification. Closure of valves 07SOV-104A, B, and C could also be accomplished by an alternative method but this method would require manual extraction and retraction of the TIP from the shield block. This method of testing would be contrary to the principles of keeping radiation exposure as low as reasonably achievable because it would result in radiation exposure to personnel performing the test.

VRR-001 TIP CIV STROKE TIME

REASON FOR REQUEST: (Continued)

The proposed alternative test ensures the operation of valves 07SOV-104A, B, and C in both directions and provides an acceptable level of quality and safety. This method meets the desired outcome of monitoring valve stroke time for degradation since the computer controls the 10-second delay and the additional approximate 2 seconds for valve closure should indicate the actual stroke time.

PROPOSED ALTERNATIVE AND BASIS:

JAF proposes to measure overall cycle time (opened and closed) for the TIP Containment Isolation Valves, 07SOV-104A, B, and C, in accordance with ISTC-5152. Specifically, JAF will time the opening (10-second delay time included) and closing cycle for valves 07SOV-104A, B, and C. The time from open initiation to receipt of the closed light for each valve will be monitored with a stop watch. JAF will apply ISTC-5152(a) which requires that each valve exhibit no more than ± 25% change in stroke time when compared to the reference value except that the full stroke limiting time for each valve will be truncated at 12 seconds.

Based on the hardship resulting from implementing the OM Code requirements without a compensating increase in the level of quality and safety, this proposed alternative is requested pursuant to 10 CFR 50.55a(z)(2).

DURATION:

5th IST Interval beginning June 1, 2018, and ending September 30, 2027

PRECEDENT:

This proposed 10 CFR 50.55a alternative was previously authorized for use at JAF under VRR-02 for the Interval 4 IST Program via the NRC safety evaluation dated November 27, 2007 (ADAMS Accession No. ML072910422).

REFERENCE:

Letter from Entergy Nuclear Operations, Inc., Pete Dietrich, to US Nuclear Regulatory Commission, dated July 31, 2007, Response to request for Additional Information Regarding Proposed Relief Requests for James A. Fitzpatrick Nuclear Power Plant Fourth Interval In-Service Testing Program (ADAMS Accession No. ML072190608).

VRR-002 EFCV SAMPLING

PLANT/UNIT:	James A. Fitzpatrick (JAF) Nuclear Power Plant
INTERVAL:	5th Interval beginning June 1, 2018, and ending September 30, 2027
COMPONENTS AFFECTED:	Excess Flow Check Valves (EFCVs) listed in the attached Table VRR-02, Excess Flow Check Valve Component List.
CODE EDITION AND ADDENDA:	ASME OM Code-2004 Edition with Addenda through OMb-2006
REQUIREMENTS:	ISTC-3522, Category C Check Valves, paragraph (c) states: “If exercising is not practicable during operation at power and cold shutdowns, it shall be performed during refueling outages.”

REASON FOR REQUEST:

Pursuant to 10 CFR 50.55a(z)(1), an alternative is proposed to the requirements of ASME OM Code paragraph ISTC-3522(c). The basis of the request is that the proposed alternative would provide an acceptable level of quality and safety. The ASME OM Code requires check valves to be exercised quarterly during plant operation, or if valve exercising is not practicable during plant operation and cold shutdown, it shall be performed during refueling outages (RFOs).

These valves cannot be exercised closed during normal power operation since closing these valves would isolate instrumentation required for power operation. Isolation of any of these instruments from service may cause a spurious signal, which could result in a plant trip or an unnecessary challenge to safety systems. Testing on a cold shutdown frequency is impractical, considering the test condition for most of the EFCVs, requires that reactor pressure is available for testing. The appropriate time for performing EFCV testing is during RFOs, and for most of the EFCVs, in conjunction with the vessel in-service leakage (hydrostatic) testing. Recent improvements in RFO schedules minimized the time that is planned for refueling and testing activities during the outages. Considering the test conditions required for EFCV testing and the shortened outage durations, it is burdensome to test all these valves during RFOs.

Based on past experience, EFCV testing during in-service leakage testing can become the outage critical path and could possibly extend the outage if all EFCVs were to be tested during this time frame.

The testing described above requires isolation of the instruments associated with each EFCV and opening of a drain valve to actuate the EFCV. Process fluid will be contaminated to some degree, requiring special measures to collect flow from the drain valve and also contributes to an increase in personnel radiation exposure.

VRR-002 EFCV SAMPLING

PROPOSED ALTERNATIVE AND BASIS:

JAF proposes to exercise test, by full-stroke to the position required to fulfill its function, a representative sample of EFCVs every refuel outage. The representative sample is based on approximately 20 percent of the valves each cycle such that each valve is tested at least once every 10 years (nominal).

Industry experience as documented in Boiling Water Reactor Owners Group (BWROG) report, NEDO-32977-A, (B21-00658-01), Excess Flow Check Valve Testing Relaxation, indicates that EFCVs have a very low failure rate. The instrument lines at JAF have a flow restricting orifice upstream of the EFCVs to limit reactor water leakage in the event of rupture. The JAF Final Safety Analysis Report (FSAR), paragraph 7.1.6, "Supplemental NSSS Supplier Information," does not credit the EFCVs, but instead credits the installed orifice for limiting the release of reactor coolant following an instrument line break. Thus, a failure of an EFCV, though not expected as a result of this request, is bounded by the FSAR analysis. The JAF test experience is consistent with the findings in the NEDO document. The NEDO document indicates similarly that many reported test failures at other plants were related to test methodologies and not actual EFCV failures.

EFCV failures will be documented in the JAF's Corrective Action Program as a surveillance test failure. The failure will be evaluated and corrected. An Equipment Failure Evaluation (EFE) will be required per the Corrective Action Program. The EFE will encompass common failure mode identification, industry experience evaluation, and review of similar component failure history.

To ensure EFCV performance remains consistent with the extended test interval, a minimum acceptance criteria of less than or equal to 1 failure per year on a 3 year rolling average will be required. Upon exceeding the criteria, a root-cause evaluation is required to determine cause, extent of conditions, an evaluation of the testing interval to ensure reliability of the EFCVs, and a risk analysis of the effects of the failures on cumulative and instantaneous plant safety. Corrective actions and performance goals will be established based on the results of the root-cause analysis.

The basis of this request is that the proposed alternative, described above, would provide an acceptable level of quality and safety. Therefore, this proposed alternative is requested pursuant to 10 CFR 50.55a(z)(1).

DURATION:

5th IST Interval, beginning June 1, 2018, and ending September 30, 2027

VRR-002 EFCV SAMPLING

PRECEDENCE:

1) This 10CFR50.55a request was previously approved as relief request VRR-03 for the JAF Interval 4 IST Program in NRC Safety Evaluation dated November 27, 2007 (ML072910422).

2) Columbia Generating Station – Requests for Relief Nos. RG01, RP01, RP02, RP03, RP04, RP05, RP06, RV01, RV02, RV03 and RV04 for the Fourth 10-Year Inservice Testing Interval, NRC Safety Evaluation dated December 09, 2014, (ML14337A449).

REFERENCES

1. BWROG Topical Report, NEDO-32977-A (B21-00658-01), "Excess Flow Check Valve Testing Relaxation," June 2000.

2. JAF FSAR, paragraph 7.1.6, "Supplemental NSSS Supplier Information"

VRR-002 EFCV SAMPLING

Table VRR-02
Excess Flow Check Valve Component List

VALVE NUMBER	SYSTEM	OM CATEGORY	ASME CLASS
02-2EFV-PS-128A	Reactor Water Recirculation	A/C	1
02-2EFV-PS-128B	Reactor Water Recirculation	A/C	1
02-2EFV-PT-24A	Reactor Water Recirculation	A/C	1
02-2EFV-PT-24B	Reactor Water Recirculation	A/C	1
02-2EFV-PT-25A	Reactor Water Recirculation	A/C	1
02-2EFV-PT-25B	Reactor Water Recirculation	A/C	1
02-2EFV1-DPT-111A	Reactor Water Recirculation	A/C	1
02-2EFV1-DPT-111B	Reactor Water Recirculation	A/C	1
02-2EFV1-FT-110A	Reactor Water Recirculation	A/C	1
02-2EFV1-FT-110C	Reactor Water Recirculation	A/C	1
02-2EFV1-FT-110E	Reactor Water Recirculation	A/C	1
02-2EFV1-FT-110G	Reactor Water Recirculation	A/C	1
02-2EFV2-DPT-111A	Reactor Water Recirculation	A/C	1
02-2EFV2-DPT-111B	Reactor Water Recirculation	A/C	1
02-2EFV2-FT-110A	Reactor Water Recirculation	A/C	1
02-2EFV2-FT-110C	Reactor Water Recirculation	A/C	1
02-2EFV2-FT-110E	Reactor Water Recirculation	A/C	1
02-2EFV2-FT-110G	Reactor Water Recirculation	A/C	1
02-3EFV-11	Nuclear Boiler	A/C	1
02-3EFV-13A	Nuclear Boiler	A/C	1
02-3EFV-13B	Nuclear Boiler	A/C	1

VRR-002 EFCV SAMPLING

Table VRR-02
Excess Flow Check Valve Component List

VALVE NUMBER	SYSTEM	OM CATEGORY	ASME CLASS
02-3EFV-15A	Nuclear Boiler	A/C	1
02-3EFV-15B	Nuclear Boiler	A/C	1
02-3EFV-15N	Nuclear Boiler	A/C	1
02-3EFV-17A	Nuclear Boiler	A/C	1
02-3EFV-17B	Nuclear Boiler	A/C	1
02-3EFV-19A	Nuclear Boiler	A/C	1
02-3EFV-19B	Nuclear Boiler	A/C	1
02-3EFV-21A	Nuclear Boiler	A/C	1
02-3EFV-21B	Nuclear Boiler	A/C	1
02-3EFV-21C	Nuclear Boiler	A/C	1
02-3EFV-21D	Nuclear Boiler	A/C	1
02-3EFV-23A	Nuclear Boiler	A/C	1
02-3EFV-23B	Nuclear Boiler	A/C	1
02-3EFV-23C	Nuclear Boiler	A/C	1
02-3EFV-23D	Nuclear Boiler	A/C	1
02-3EFV-23	Nuclear Boiler	A/C	1
02-3EFV-25	Nuclear Boiler	A/C	1
02-3EFV-31A	Nuclear Boiler	A/C	1
02-3EFV-31B	Nuclear Boiler	A/C	1
02-3EFV-31C	Nuclear Boiler	A/C	1
02-3EFV-31D	Nuclear Boiler	A/C	1
02-3EFV-31E	Nuclear Boiler	A/C	1
02-3EFV-31F	Nuclear Boiler	A/C	1

VRR-002 EFCV SAMPLING

Table VRR-02
Excess Flow Check Valve Component List

VALVE NUMBER	SYSTEM	OM CATEGORY	ASME CLASS
02-3EFV-31G	Nuclear Boiler	A/C	1
02-3EFV-31H	Nuclear Boiler	A/C	1
02-3EFV-31J	Nuclear Boiler	A/C	1
02-3EFV-31K	Nuclear Boiler	A/C	1
02-3EFV-31L	Nuclear Boiler	A/C	1
02-3EFV-31JM	Nuclear Boiler	A/C	1
02-3EFV-31N	Nuclear Boiler	A/C	1
02-3EFV-31P	Nuclear Boiler	A/C	1
02-3EFV-31R	Nuclear Boiler	A/C	1
02-3EFV-31S	Nuclear Boiler	A/C	1
02-3EFV-33	Nuclear Boiler	A/C	1
13EFV-01A	Nuclear Boiler	A/C	1
13EFV-01B	Nuclear Boiler	A/C	1
13EFV-02A	Nuclear Boiler	A/C	1
13EFV-02B	Nuclear Boiler	A/C	1
14EFV-31A	Core Spray	A/C	1
14EFV-31B	Core Spray	A/C	1
23EFV-01A	High Pressure Coolant Injection	A/C	1
23EFV-01B	High Pressure Coolant Injection	A/C	1
23EFV-02A	High Pressure Coolant Injection	A/C	1
23EFV-02B	High Pressure Coolant Injection	A/C	1
29EFV-30A	Main Steam	A/C	1
29EFV-30B	Main Steam	A/C	1

VRR-002 EFCV SAMPLING

Table VRR-02

Excess Flow Check Valve Component List

VALVE NUMBER	SYSTEM	OM CATEGORY	ASME CLASS
29EFV-30C	Main Steam	A/C	1
29EFV-30D	Main Steam	A/C	1
29EFV-34A	Main Steam	A/C	1
29EFV-34B	Main Steam	A/C	1
29EFV-34C	Main Steam	A/C	1
29EFV-34D	Main Steam	A/C	1
29EFV-53A	Main Steam	A/C	1
29EFV-53B	Main Steam	A/C	1
29EFV-53C	Main Steam	A/C	1
29EFV-53D	Main Steam	A/C	1
29EFV-54A	Main Steam	A/C	1
29EFV-54B	Main Steam	A/C	1
29EFV-54C	Main Steam	A/C	1
29EFV-54D	Main Steam	A/C	1

VRR-003 CODE CASE OMN-17

PLANT/UNIT:	James A. Fitzpatrick (JAF) Nuclear Power Plant
INTERVAL:	5th Interval beginning June 1, 2018, and ending September 30, 2027
COMPONENTS AFFECTED:	Main Steam Safety/Relief Valves (SRVs), 02RV-071A through 02RV-071H and 02RV-071J through 02RV-071L
CODE EDITION AND ADDENDA:	ASME OM Code-2004 Edition with Addenda through OMb-2006
REQUIREMENTS:	ASME Code Mandatory Appendix I, paragraph I-1320(a), Test Frequencies, Class 1 Pressure Relief Valves, states, in part, that "Class 1 pressure relief valves shall be tested at least once every 5 years …"
REASON FOR REQUEST:	Pursuant to 10 CFR 50.55a(z)(1), an alternative is requested to the frequency requirement for testing Class 1 pressure relief valves every 5 years. JAF proposes to test Class 1 pressure relief valves every 6 years in accordance with ASME Code Case OMN-17, Alternative Rules for Testing ASME Class 1 Pressure Relief/Safety Valves, which was published via the American Society of Mechanical Engineers (ASME) Operation and Maintenance of Nuclear Power Plants (OM) Code 2009 Edition. JAF considers that testing the main steam SRVs in accordance with ASME OM Code Case OMN-17 provides an acceptable level of quality and safety.
PROPOSED ALTERNATIVE AND BASIS:	ASME OM Code Mandatory Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants," paragraph I-1320, "Test Frequencies, Class 1 Pressure Relief Valves," (a) "*5-Year Test Interval*," requires that Class 1 pressure relief valves be tested at least once every 5 years. As an alternative to the Code required 5-year test interval per Mandatory Appendix I, paragraph I-1320(a), JAF proposes that Class 1 pressure relief valves, 02RV-071A, B, C, D, E, F, G, H, J, K, and L, be tested at least once every three (3) refueling cycles (approximately 6 years/72 months) with a minimum of 20% of the valves tested within any 24-month interval. This 20% would consist of valves (complete assemblies) that have not been tested during the current 72-month interval, if they exist. The test interval for any individual valve would not exceed 72 months except that a 6-month grace period is allowed to coincide with refueling outages to accommodate extended shutdown periods. JAF proposes to continue testing all eleven (11) installed pilot valves every refueling outage. The main steam pressure relief system provides reactor coolant system (RCS) overpressure protection and automatic depressurization of the nuclear system by opening the SRVs. JAF updated Final Safety Analysis Report Section 4.4.4, "Safety Design Bases," describes the main protection functions of the SRVs as follows:

VRR-003 CODE CASE OMN-17

PROPOSED ALTERNATIVE AND BASIS (Continued):

1. The pressure relief system prevents over-pressurization of the RCS in order to prevent failure of the reactor coolant pressure boundary due to pressure.

2. The pressure relief system provides automatic depressurization for small breaks in the RCS so that the low pressure coolant injection and the core spray systems can operate to protect the fuel barrier.
3. The pressure relief system provides for manual depressurization at a remote auxiliary panel located outside the control room in the highly unlikely event the control room was to become uninhabitable.

Seven (7) of the eleven (11) SRVs are a designated part of the automatic depressurization system (ADS) emergency core cooling system (ECCS) and must open to provide automatic reactor depressurization as a result of a small break in the nuclear system for which the high pressure coolant injection system cannot maintain reactor water level (ADS function).

The relief valve testing and maintenance cycle at JAF consists of an as-found inspection, seat leakage and set pressure testing. After as-found set pressure testing, the valves shall be disassembled and inspected to verify that parts are free of defects resulting from time-related degradation or service induced wear. As-left set pressure testing shall be performed following maintenance and prior to returning the valve to service.

Prior to the return of a complement of SRVs for installation in the plant, the valves are disassembled and inspected to verify that internal surfaces and parts are free from defects or service induced wear prior to the start of the next test interval. During this process, anomalies or damage are identified and resolved. Damaged or worn parts, springs, gaskets and seals are replaced as necessary. Following reassembly, the valve's set pressure is recertified with an acceptance criterion of ±1%. This existing process is in accordance with ASME OM Code Case OMN-17, paragraphs (d) and (e). This Code Case has not been approved for use in US NRC Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code, dated June 2003.

The JAF SRV pilot valves' test history, as described in Attachment 1, Safety Relief Valve Pilot Valve Test Data, shows that the pilot valves are susceptible to disc-to-seat corrosion. This is a recurring industry issue that has been the subject of both NRC and Boiling Water Reactor Owner's Group (BWROG) assessments. As described in Licensee Event Report (LER) 2015-002-00, dated July 30, 2015, and based on the known industry wide issues with the two-stage Target Rock SRVs, JAF has implemented the following industry recommendations:

VRR-003 CODE CASE OMN-17

PROPOSED ALTERNATIVE AND BASIS (Continued):

1. Installed Stellite 21 discs in the SRV pilot assemblies during refurbishment;
2. Installed the electric lift system recommended by the BWROG;
3. Installed enhanced insulation on the SRVs;
4. Redirected ventilation air flow away from the SRVs; and
5. Began a phased replacement of 2-stage SRVs with 3-stage.

As a result, JAF will continue to test and inspect 100% of the installed SRV pilot valves each refueling outage until such time that successful test performance is achieved. The inspection and testing of the SRV valve assemblies, performed to date, have not identified any relevant issues.

However, given the current 24-month operating cycle, JAF would be required to remove and test fifty (50) percent of the SRVs every refueling outage (i.e. five or six of 11), such that all valves are removed and tested every two refueling outages. This would ensure compliance with the ASME OM Code requirements for testing Class 1 pressure relief valves within a five-year interval. Approval of extending the test interval for the valves to 6 years with a grace period of 6 months, consistent with Code Case OMN-17, would reduce the minimum number of SRVs tested at JAF over three refueling outages by a minimum of five.

Pursuant to 10 CFR 50.55a(z)(1), an alternative is proposed to the frequency requirement for testing Class 1 pressure relief valves every 5 years. JAF proposes to test Class 1 pressure relief valves every 6 years, with a grace period of six months, in accordance with ASME Code Case OMN-17, Alternative Rules for Testing ASME Class1 Pressure Relief/Safety Valves. JAF proposes to continue testing all eleven (11) installed pilot valves every refueling outage until such time that successful test performance is achieved. JAF considers that testing the main steam SRVs in accordance with ASME OM Code Case OMN-17 and the installed SRV pilot valves each refueling outage will provide an acceptable level of quality and safety.

DURATION: 5th IST Interval, beginning June 1, 2018, and ending September 30, 2027

PRECEDENTS:

1. Approved for JAF Interval 4 IST Program, as relief request VRR-06, in NRC Safety Evaluation dated October 1, 2009 (ML092730032).

2. Approved for Entergy Operation, Incorporated, in NRC SER dated March 23, 2015, River Bend Station, Unit 1 - Relief Request No. VRR-RBS-2014-1 Regarding the Third 10-year Inservice Test Interval (TAC No, MF4125), (ML15071A141).

VRR-003 CODE CASE OMN-17

REFERENCES:

1. JAF LER 2015-002-00, Safety Relief Valve Upward Setpoint Drift, dated July 30, 2015, (ML15212A272).

2. ASME OM Code Case OMN-17, Alternative Rules for Testing ASME Class 1 Pressure Relief/Safety Valves

3. JAF FSAR Update, Revision 5, Section 4.4, Pressure Relief System, subsection 4.4.4, Safety Design Basis

VRR-003 CODE CASE OMN-17

Attachment 1, Safety Relief Valve Pilot Valve Test Data

S/N	Cycle 18 Installed Location	Model	As Found Set Pressure	Deviation (%)	Set Pressure
1047	02RV-71C	2-stage	1184	3.29%	1145
1080	02RV-71E	2-stage	1187	3.54%	1145
1052	02RV-71F	2-stage	No Lift		1145
1088	02RV-71A	2-stage	1148	0.26%	1145
1237	02RV-71B	2-stage	1192	3.94%	1145
1111	02RV-71D	2-stage	1157	1.04%	1145
1053	02RV-71G	2-stage	No Lift		1145
1194	02RV-71H	2-stage	1169	2.05%	1145
1192	02RV-71K	2-stage	1156	0.95%	1145
1056	02RV-71J	2-stage	1168	1.97%	1145
1053	02RV-71L	2-stage	1177	2.72%	1145
1193		2-stage	1245	8.03%	1145
1238			1227	6.68%	1145

S/N	Cycle 19 Installed Location	Model	As Found Set Pressure	Deviation (%)	Set Pressure
1045	02RV-71C	3-stage	1206	5.33%	1145
1191	02RV-71E	3-stage	No Lift		1145
1236	02RV-71F	3-stage	No Lift		1145
1087	02RV-71A	2-stage	1163	N/A	1145
1218	02RV-71B	2-stage	1208	5.50%	1145
1143	02RV-71D	2-stage			1145
1217	02RV-71G	2-stage	1192	4.10%	1145
1235	02RV-71H	2-stage	No Lift		1145
1051	02RV-71K	2-stage	No Lift		1145
1195	02RV-71J	2-stage	1246	8.82%	1145
1052	02RV-71L	2-stage	1163	1.57%	1145
1238			1189	3.84%	1145
1013		2-stage	1159	1.22%	1145
1110		2-stage	1164	1.66%	1145

VRR-003 CODE CASE OMN-17

Attachment 1, Safety Relief Valve Pilot Valve Test Data (Continued)

S/N	Cycle 20 Installed Location	Model	As Found Set Pressure	Deviation (%)	Set Pressure
TR2	02RV-71C	3-stage	1132	-1.14%	1145
TR3	02RV-71E	3-stage	1155	0.87%	1145
TR4	02RV-71F	3-stage	1151	0.52%	1145
1088	02RV-71A	2-stage	1195	4.37%	1145
1193	02RV-71B	2-stage	1150	0.44%	1145
1194	02RV-71D	2-stage	1159	1.22%	1145
1047	02RV-71G	2-stage	1150	0.44%	1145
1192	02RV-71H	2-stage	1183	3.32%	1145
1111	02RV-71K	2-stage	1162	1.48%	1145
1080	02RV-71J	2-stage	1177	2.79%	1145
1056	02RV-71L	2-stage	1141	-0.35%	1145

S/N	Cycle 21 Installed Location	Model	As Found Set Pressure	Deviation (%)	Set Pressure
11	02RV-71F	3-stage	1133	-1.05%	1145
58	02RV-71E	3-stage	1118	-2.36%	1145
61	02RV-71C	3-stage	1122	-2.01%	1145
1013	02RV-71A	2-stage	1140	-0.44%	1145
1045	02RV-71B	2-stage	1250	9.17%	1145
1052	02RV-71D	2-stage	1191	4.02%	1145
1110	02RV-71G	2-stage	1231	7.51%	1145
1195	02RV-71H	2-stage	1235	7.86%	1145
1217	02RV-71K	2-stage	1243	8.56%	1145
1218	02RV-71J	2-stage	1277	11.53%	1145
1235	02RV-71L	2-stage	1203	5.07%	1145

VRR-04 Leak Test At APP J Frequency

PLANT/UNIT:	James A. Fitzpatrick (JAF) Nuclear Power Plant
INTERVAL:	5th Interval beginning June 1, 2018, and ending September 30, 2027
COMPONENTS AFFECTED:	Pressure Isolation Valves (PIVs) listed in the attached Table VRR-04-01, Pressure Isolation Valve Component List.
CODE EDITION AND ADDENDA:	ASME OM Code-2004 Edition with Addenda through OMb-2006
REQUIREMENTS:	ISTC-3630, “Leakage Rate for Other Than Containment Isolation Valves”, paragraph ISTC-3630(a), *Frequency*, states: "Tests shall be conducted at least once every 2 years.”
REASON FOR REQUEST:	Pursuant to 10 CFR 50.55a(z)(1), an alternative is proposed to the frequency requirements of ASME OM Code ISTC-3630(a) for the subject valves. ISTC-3630(a) requires that leakage rate testing of Category A valves with a leakage requirement that is not based on 10CFR50 Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” be conducted at least every two years. At JAF, PIVs are Category A or Category A/C valves within the scope of ISTC-3630. This alternative, to allow for scheduling of leak tests, for the valves identified in Table VRR-04-01, to a performance-based frequency that is the same as 10 CFR 50, Appendix J, Option B, “Performance-Based Requirements,” testing, would provide an acceptable level of quality and safety.
PROPOSED ALTERNATIVE AND BASIS:	JAF proposes an alternative test frequency in lieu of the requirements found in ISTC-3630(a) for the 10 applicable PIVs listed in the attached Table VRR-04-01. Specifically, JAF proposes to perform the valve leakage rate test at a frequency in accordance with the 10 CFR 50, Appendix J, Option B schedule. The identified valves were initially tested at the ISTC-3630(a) required interval schedule, which was every refueling outage (RFO) or 2 years, during the 3rd ten-year IST interval at JAF, and have subsequently been tested at the 10 CFR 50, Appendix J, Option B frequency, if applicable, during the 4th ten-year IST interval. Valves that demonstrated good performance for two consecutive cycles may have had their test interval extended to a maximum of 60 months. Any leakage rate test failure required the component to return to the initial interval of every RFO or 2 years until good performance could again be established. 10 CFR 50, Appendix J, was amended to improve the focus of the body of regulations by eliminating prescriptive requirements that are marginal to safety and to provide licensees greater flexibility for cost-effective implementation methods for meeting regulatory safety objectives. Consistent with this approach, 10CFR50, Appendix J, Option B is less prescriptive, utilizes risk-based insights and allows licensees to adopt

VRR-04 Leak Test At APP J Frequency

PROPOSED ALTERNATIVE AND BASIS (Continued):

cost effective methods, including setting test intervals, for implementing safety objectives underlying the requirements of Appendix J.

Nuclear Energy Institute (NEI) 94-01, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, as modified by the positions in Regulatory Guide 1.163, describes the risk-informed basis for extended test intervals under Option B. That justification documents valves which have demonstrated good leakage rate performance over two consecutive cycles are subject to future failures predominantly governed by the random failure rate of the component. NEI 94-01 also presents the results of a comprehensive risk analysis, including the statement that "the risk impact associated with increasing test intervals is negligible (less than 0.1 percent of total risk)".

NUREG 0933, “Resolution of Generic Safety Issues,” Issue 105, “Interfacing Systems LOCA at LWRs,” discussed the need for PIV leak rate testing based primarily on three pre-1980 historical failures of applicable valves industry-wide. These failures all involved human errors in either operations or maintenance. None of these failures involved inservice equipment degradation. The performance of PIV leak rate testing provides assurance of acceptable seat leakage with the valve in a closed condition. Typical PIV testing does not identify functional problems which may inhibit the valves ability to re-position from open to close.

For check valves, such functional testing is accomplished per ASME OM Code ISTC-3520, “Exercising Requirements,” and ISTC-3522, “Category C Check Valves”. Power-operated valves are routinely full stroke tested per ASME OM Code to ensure their functional capabilities. The functional tests for PIVs are performed only at an RFO frequency. Such testing is not performed online in order to prevent any possibility of an inadvertent Interfacing System Loss of Coolant Accident (ISLOCA) condition. The functional testing of the PIVs is adequate to identify any abnormal condition that might affect closure capability. Performance of the separate PIV leak rate testing does not contribute any additional assurance of functional capability; it verifies the seat tightness of the closed valves. JAF proposes to perform PIV testing at intervals specified in NEI 94-01. The specific interval for each valve would be a function of its performance and would be established in a manner consistent with the Containment Isolation valve (CIV) process under 10 CFR 50, Appendix J, Option B. Program guidance will be established such that if any of the valves fail either the CIV test or PIV test, the test interval for both tests will be reduced to once every 30 months until they can be re-classified as good performers per the performance evaluation requirements of Appendix J, Option B. The test intervals for the valves identified in this request will be determined in the same manner as is done for CIV testing under Option B. That is, the test interval may be extended upon completion of two consecutive periodic PIV tests with results within the

VRR-04 Leak Test At APP J Frequency

PROPOSED ALTERNATIVE AND BASIS (Continued):

prescribed acceptance criteria. Any PIV test failure will require a return to the initial interval until good performance can again be established.

The intent of this request is to allow for a performance-based approach to the scheduling of PIV leakage testing. It has been shown that ISLOCA represents a small risk impact to BWRs such as JAF. NUREG 0933 Issue 105, references the conclusion from NUREG/CR-5928, "Final Report of the NRC-sponsored ISLOCA Research Program," that concludes, for the units studied, ISLOCA poses little risk.

The PIVs have an excellent performance history in terms of seat leakage testing (See Attachment 1, Pressure Isolation Valve Component Leakage History). It should be noted that component tests identified in Attachment 1 as “LJ-C” are performed using air as the test media. Alternatively, component tests identified in Attachment 1 as “LKO” are performed using water as the test media.

The risks associated with extending the leakage test interval to a maximum of sixty months are extremely low. The basis for this alternative request is the historically good performance of the PIVs. This alternative will also provide significant reductions in radiation dose. Approximately 500 mRem per outage is received when a full complement of PIV tests is required. The last RFO radiation exposure for which a full complement of PIVs was tested was used to identify that PIV testing alone each RFO resulted in a total dose to personnel of approximately 500 mRem.

It should be noted that JAF submitted a license amendment request (LAR) on August 29, 2016. The LAR proposes to change the plant Technical Specification (TS) 5.5.6 by replacing the reference to Regulatory Guide 1.163 with a reference to the NEI 94-01, Revision 3-A, topical report, dated July 2, 2012. This proposed change will allow an extension from the 60-month frequency currently permitted by Option B to a 75-Month frequency for type C leakage rate testing. Upon approval of this LAR, JAF will implement the 75-month testing frequency for those PIVs that are CIV’s as listed in this alternative request and which demonstrate acceptable performance over two consecutive cycles as defined by the measured leak rate being less than the leak rate acceptance criteria for PIV and CIV testing.

Based on the determination that the scheduling of leak tests, for the valves identified in this request, to a performance based frequency that is the same as 10 CFR 50, Appendix J, Option B testing, would provide an acceptable level of quality and safety, this proposed alternative is requested pursuant to 10 CFR 50.55a(z)(1).

DURATION: 5th IST Interval, beginning June 1, 2018, and ending September 30, 2027

PRECEDENT: This proposed 10 CFR 50.55a alternative was previously authorized for use at JAF under VRR-07 for the Interval 4 IST Program in NRC Safety Evaluation dated March 16, 2012 (ML12072A113).

VRR-04 Leak Test At APP J Frequency

REFERENCES:

1. NUREG 0933, “Resolution of Generic Safety Issues,” Issue 105, “Interfacing Systems LOCA at LWRs”
2. NUREG/CR-5928, ISLOCA Research Program, (ML072430731)
3. 10 CFR 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors”
4. NEI 94-01, Rev 3-A, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J”
5. RG 1.163, “Performance-Based Containment Leak-Test Program," dated September 1995 (ML003740058).
6. LAR submitted to NRC August 2016 to incorporate reference to NEI 94-01, Rev. 3-A

VRR-04 Leak Test At APP J Frequency

Table VRR-04-01
Pressure Isolation Valve Component List

VALVE NUMBER	DESCRIPTION	APPENDIX J OPTION B AIR TESTED	OM CATEGORY
10AOV-68A	RHR A LPCI TESTABLE CHECK VALVE	No	A/C
10AOV-68B	RHR B LPCI TESTABLE CHECK VALVE	No	A/C
10MOV-25A	RHR A LPCI INBD INJ VALVE	Yes	A
10MOV-25B	RHR B LPCI INBD INJ VALVE	Yes	A
14AOV-13A	CSP A REACTOR ISOL TESTABLE CHECK VALVE	No	A/C
14AOV-13B	CSP B REACTOR ISOL TESTABLE CHECK VALVE	No	A/C
14MOV-12A	CORE SPRAY LOOP A INBD ISOL VALVE	Yes	A
14MOV-12B	CORE SPRAY LOOP B INBD ISOL VALVE	Yes	A
10MOV-17	RHR SHUTDOWN COOLING OUTBD ISOL VALVE	Yes	A
10MOV-18	RHR SHUTDOWN COOLING INBD ISOL VALVE	Yes	A

VRR-04 Leak Test At APP J Frequency

Attachment 1 - Pressure Isolation Valve Component Leakage History

10AOV-68A Leakage History

Test Date	Test Type	Test Result	Leakage	Leakage Limit	Leakage Units
10/10/2002	LKO	Sat	0.5	10	GPM
10/9/2004	LKO	Sat	0	10	GPM
10/29/2006	LKO	Sat	0	10	GPM
9/29/2008	LKO	Sat	0	10	GPM
9/19/2010	LKO	Sat	0	10	GPM
9/25/2012	LKO	Sat	0	10	GPM
8/30/2014	LKO	Sat	0	10	GPM

10AOV-68B Leakage History

Test Date	Test Type	Test Result	Leakage	Leakage Limit	Leakage Units
10/17/2002	LKO	Sat	0	10	GPM
9/29/2004	LKO	Sat	8.5	10	GPM
10/29/2006	LKO	Sat	0	10	GPM
9/21/2008	LKO	Sat	0	10	GPM
9/29/2010	LKO	Sat	*[1]	10	GPM
9/21/2012	LKO	Sat	0.11	10	GPM
9/12/2014	LKO	Sat	0	10	GPM

*[1] – Documentation lost and test results not recorded

VRR-04 Leak Test At APP J Frequency

Attachment 1 - Pressure Isolation Valve Component Leakage History (Continued)

10MOV-25A Leakage History

Test Date	Test Type	Test Result	Leakage	Leakage Limit	Leakage Units
10/10/2004	LJ-C	Sat	1.184	16	SLM
9/30/2008	LJ-C	Sat	5.96	5.893	SLM
9/25/2012	LJ-C	Sat	0.35	5.893	SLM
8/30/2014	LJ-C	Sat	0.45	5.893	SLM
9/3/2002	LKO	Sat	0	5	GPM
8/30/2006	LKO	Sat	0	5	GPM
9/10/2010	LKO	Sat	0.24	5	GPM
6/21/2012	LKO	Sat	0	5	GPM
8/10/2016	LKO	Sat	0	5	GPM

10MOV-25B Leakage History

Test Date	Test Type	Test Result	Leakage	Leakage Limit	Leakage Units
10/5/2004	LJ-C	Sat	3.85	5.893	SLM
10/26/2006	LJ-C	Sat	8.85	5.893	SLM
9/16/2008	LJ-C	Sat	1.53	16	SLM
9/23/2008	LJ-C	Sat	4.88	5.893	SLM
9/23/2012	LJ-C	Sat	0.62	5.893	SLM
9/29/2012	LJ-C	Sat	10.4	5.893	SLM
9/3/2002	LKO	Sat	0	10	GPM
9/3/2006	LKO	Sat	0	10	GPM
8/29/2010	LKO	Sat	0	10	GPM
11/20/2016	LKO	Sat	0	10	GPM

VRR-04 Leak Test At APP J Frequency

Attachment 1 - Pressure Isolation Valve Component Leakage History (Continued)

14AOV-13A Leakage History

Test Date	Test Type	Test Result	Leakage	Leakage Limit	Leakage Units
10/8/2002	LKO	Sat	0	10	GPM
10/9/2004	LKO	Sat	0	10	GPM
10/14/2006	LKO	Sat	0	10	GPM
9/22/2008	LKO	Sat	0	10	GPM
9/20/2010	LKO	Sat	0	10	GPM
10/5/2012	LKO	Sat	0	10	GPM
8/26/2014	LKO	Sat	0	10	GPM

ATTACHMENT 6

RELIEF REQUEST RAIs AND SERs

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 13, 2018

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - ISSUANCE OF RELIEF REQUESTS FOR ALTERNATIVES TO CERTAIN ASME 0M CODE REQUIREMENTS (CAC NOS. MG0052, MG0053, MG0054, MG0055, MG0056, MG0057, MG0058, AND MG0061•, EPID L-2017-LLR-0067, L-2017-LLR-0068, L-2017-LLR-0069, L-2017-LLR-0070, L-2017-LLR-0071 , L-2017-LLR-0072, L-2017-LLR-0073, AND 1-2017-LLR-0074)

Dear Mr. Hanson:

By letter dated August 3, 2017, as supplemented by letter dated October 26, 2017 (Agency wide Documents Access and Management System (ADAMS) Accession Nos. ML 17219A123 and ML 17299A560, respectively), Exelon Generation Company, LLC (Exelon or the licensee) submitted eight relief requests to the U.S. Nuclear Regulatory Commission (NRC) for relief from certain requirements of the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (0M Code), 2004 Edition with Addenda through Omb-2006, at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick) associated with the fifth 10-year inservice testing interval.

Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety for Relief Requests PRR-01 , PRR-02, PRR-04, VRR-02, VRR-03, and VRR-04. Pursuant to 10 CFR 50.55a(z)(2) the licensee requested to use the proposed alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety for Relief Requests PRR-03 and VRR-01

The NRC staff has reviewed the subject requests and concludes, as set forth in the enclosed safety evaluations, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) and 10 CFR 50.55a(z)(2).

The NRC staff has determined that for requests PRR-01 , PRR-02, PRR-04, VRR-02, VRR-03, and VRR-04, the proposed alternatives provide an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) for these proposed alternatives. Therefore, the NRC staff authorizes the use of the proposed alternatives for requests PRR-01, PRR-02, PRR-04, VRR-02, VRR-03, and VRR-04 for FitzPatrick for the fifth 10-year inservice testing interval at FitzPatrick, which is scheduled to begin on June 1, 2018, and end on September 30, 2027.

The NRC staff has determined that for requests PRR-03 and VRR-01, the proposed alternatives provide reasonable assurance that the affected components are operationally ready. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2) for these requests. Therefore, the NRC staff authorizes the use of the proposed alternatives for requests PRR-03 and VRR-01 for FitzPatrick for the fifth 10-year inservice testing interval at FitzPatrick, which is scheduled to begin on June 1, 2018, and end on September 30, 2027.

All other ASME 0M Code requirements for which relief was not specifically requested and approved in the subject requests remain applicable.

If you have any questions, please contact Tanya Hood at 301-415-1387 or Tanya.Hood@nrc.gov.

Sincerely,

James G. Danna. Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures:
1. Safety Evaluation for PRR-01 and PRR-04
2. Safety Evaluation for PRR-02 and PRR-03
3. Safety Evaluation for VRR-03
4. Safety Evaluation for VRR-01, VRR-02, and VRR-04

cc: Listserv

UNITED STATES

NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUESTS PRR-01, REVISION 0, AND PRR-04, REVISION 0

FIFTH 10-YEAR INTERVAL INSERVICE TESTING PROGRAM

EXELON GENERATING COMPANY, LLC

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1 .0 INTRODUCTION

By letter dated August 3, 2017, as supplemented by letter dated October 26, 2017 Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML 17219A123 and ML 17299A560, respectively), Exelon Generation Company, LLC (Exelon or the licensee) submitted Relief Requests PRR-01, Revision 0, and PRR-04, Revision 0, to the U.S. Nuclear Regulatory Commission (NRC or the Commission). The licensee requested alternative tests in lieu of certain inservice testing (IST) requirements of the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (0M Code) for the IST program at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick) during the fifth 10-year IST program interval, which is scheduled to begin on June 1, 2018, and end on September 30, 2027. Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative in Relief Requests PRR-01 and PRR-04 on the basis that the alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Section 50.55a(f), "Preservice and Inservice testing requirements," of 10 CFR requires, in part, that IST of certain ASME Code Class 1, 2, and 3 components must meet the requirements of the ASME 0M Code and applicable addenda, except where alternatives have been authorized pursuant to paragraphs 10 CFR 50.55a(z)(1) or 10 CFR 50.55a(z)(2).

Section 50.55a(z) of 10 CFR requires that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a, or portions thereof, must be submitted and authorized by the NRC prior to implementation. The applicant or licensee must demonstrate that: (1) the proposed alternative would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety. Section 50.55a of 10 CFR allows the NRC to authorize alternatives and to grant relief from ASME Code requirements upon making the necessary findings.

Enclosure 1

The guidance that the NRC staff considered in its review is NUREG-1482, Revision 2, "Guidelines for Inservice Testing at Nuclear Power Plants," October 2013 (ADAMS Accession No. ML 13295A020), which provides acceptable guidance to licensees to establish a basic understanding of the regulatory basis for pump and valve IST programs.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

The applicable ASME 0M Code edition and addenda for FitzPatrick during the fifth 10-year IST program interval is the 2004 Edition through the 2006 Addenda.

3.1 Licensee's Relief Request PRR-01, Revision O

Applicable Code Requirements

Subsection IST B-3510(a), "Accuracy," states, in part, "Instrument accuracy shall be within the limits of Table ISTB-3510-1 ."

Subsection ISTB-3510(b), "Range," (1) states, "The full-scale range of each analog instrument shall be not greater than three times the reference value."

Table IST B-3510-1 , "Required Instrument Accuracy," requires pressure and differential pressure accuracy to be ± 2 percent for pump Group A and Group B tests.

Components for Which Relief is Requested

Core Spray Pump 14P-1A
Core Spray Pump 14P-1B

Reason for Alternative Request

The differential pressure for each core spray pump is calculated using the installed suction and discharge pressure gauges. The suction pressure gauge is designed to provide adequate suction pressure indication during all expected operating conditions. The suction pressure gauge full-scale range, 72.28 pounds per square inch gauge (psig), is sufficient for a post-accident condition when the torus is at the maximum accident pressure. This range exceeds the range limit of 15 psig for the suction pressure under the test condition (approximately 5 psig).

The installed suction pressure gauge and discharge pressure instrumentation loop are calibrated to within ± 2 percent of full scale. However, the installed suction pressure instrument loop is calibrated to within ± 2.4 percent full-scale accuracy. The full-scale range of the pump discharge pressure instrumentation loop is 500 psig. Pump discharge pressure during testing is typically 280 psig. The maximum variation due to inaccuracy in measured suction pressure is ± 1.7 psi. The maximum variation due to inaccuracy in measured discharge pressure is ± 10 psi. Thus, the differential pressure (typical [Pdisch - Psuct]) would be 280 — 5, which is 275 ± 11.7 psi, for an inaccuracy of 4.3 percent. If the full-scale range of the suction pressure gauge is within the ASME 0M Code allowable value of three times the reference value

(15 psig), the maximum variation due to inaccuracy in measured suction pressure would be ± 0.3 psi, and the resulting differential pressure measurement would be 275 ± 10.3 psi, or an inaccuracy of 3.7 percent.

The decrease in inaccuracy of 0.6 percent by using the ASME 0M Code-compliant gauge is insignificant and does not warrant manpower and exposure to change the suction pressure quarterly for test purposes.

In addition, the ASME 0M Code would allow a full-scale range for the discharge pressure measurement of 840 psig for the typical 280 psig discharge pressure. This would translate into a differential pressure measurement of 275 ± 17.1 psig, or an inaccuracy of 6.2 percent, if the installed instrument met the ASME 0M Code requirements of O - 15 psig for suction pressure gauge and O - 840 psig for the discharge pressure gauge. The existing measurement is significantly better than the maximum ASME 0M Code allowable inaccuracy.

Proposed Alternative Testinq

The licensee stated that the existing installed plant suction pressure and discharge gauges will be used to determine the pump differential pressure for the core spray pumps 14P-1A and 14P-1B Group B pump tests. The comprehensive tests are performed using measuring and test equipment.

NRC Staff Evaluation

The NRC staff reviewed the information in Relief Request PRR-01, Revision O, including the supplemental information in the October 26, 2017, submittal. For Group A and Group B tests, the ASME 0M Code requires instrument accuracy to be within ± 2 percent of full scale and the full scale range of each instrument be no greater than three times the reference value. The combination of these two requirements results in an effective accuracy requirement of approximately ±6 percent of the reference value.

Based on its review, the NRC staff finds that the maximum inaccuracy of the installed suction and discharge pressure instruments combined is ± 7.3 psig [(± 1.73 psig) + (± 5.6 psig)]. The accuracies of the installed core spray pump suction and discharge instruments yield differential pressure readings that are more accurate than the readings achieved from instruments that meet the ASME 0M Code requirements [(± 16.8 psig) + (± 0.3 psig), or ± 17.1 psig] and, thus, provide an acceptable level of quality and safety.

The use of the existing pressure instrument is supported by NUREG-1482, Revision 2, paragraph 5.5.1 , which states that that the NRC staff may authorize an alternative when the combination of range and accuracy yields a reading at least equivalent to the reading achieved from the instrument that meets the ASME 0M Code requirements. Therefore, the NRC staff concludes that the proposed alternative provides an acceptable level of quality and safety.

3.2 Licensee's Relief Request PRR-04

Applicable Code Requirements

Subsection IST B-5121 , "Group A Test Procedure," (b) states, in part, "The resistance of the system shall be varied until the flow rate equals the reference point. Alternatively, the flow rate shall be varied until the differential pressure equals the reference point..."

Subsection ISTB-5122, "Group B Test Procedure," (c), states, "System resistance may be varied as necessary to achieve the reference point."

Subsection ISTB-5123, "Comprehensive Test Procedure," (b), states, in part, "the resistance of the system shall be varied until the flow rate equals the reference point. Alternatively, the flow rate shall be varied until the differential pressure equals the reference point..."

Subsection ISTB-5221, "Group A Test Procedure," (b), states, in part, "The resistance of the system shall be varied until the flow rate equals the reference point. Alternatively, the flow rate shall be varied until the differential pressure equals the reference point... "

Subsection ISTB-5222, "Group B Test Procedure," (c), states, "System resistance may be varied as necessary to achieve the reference point."

Subsection ISTB-5223, "Comprehensive Test Procedure," paragraph IST B-5123(b) states, in part, "The resistance of the system shall be varied until the flow rate equals the reference point, Alternatively, the flow rate shall be varied until the differential pressure equals the reference point..."

Components for Which Relief is Requested

Table 1

Pump ID	Pump Description	Pump Type	Code Class	0M Code Category
11P-2A 11P-2B	Standby Liquid Control Pumps	Positive Displacement	2	Group B
14P-1A 14P-1B	Core Spray Pumps	Centrifugal	2	Group B
10P-3A 10P-3B 10P-3C 10P-3D	Residual Heat Removal (RHR)/Low Pressure Coolant Injection (LPCI) Pumps	Centrifugal	2	Group A
23P-1M	High Pressure Coolant Injection (HPCI) Main Pump	Centrifugal	2	Group B
23P-1B	HPCI Booster Pump	Centrifugal	2	Group B
10P-1A 10P-1B 10P-1C 10P-1D	RHR Service Water Pump	Vertical Line Shaft Centrifugal	3	Group A
46P-2A	Emergency Service Water Pumps	Vertical Line Shaft Centrifugal	3	Group B

Reason for Alternative Request

For pump testing, there is difficulty adjusting system throttle valves with sufficient precision to achieve an exact flow rate, differential pressure, or discharge pressure during subsequent IST tests. Subsection ISTB of the ASME 0M Code does not allow for variance from a fixed reference value for pump testing. However, NUREG-1482, Revision 2, Section 5.3, acknowledges that certain pump system designs do not allow for the licensee to set the flow at an exact value because of limitations in the instruments and controls for maintaining steady flow.

The ASME 0M Code Case OMN-21, "Alternative Requirements for Adjusting Hydraulic Parameters to Specified Reference Points," provides guidance for adjusting reference flow, differential pressure (AP), or discharge pressure to within a specified tolerance during pump inservice testing. The ASME 0M Code Case OMN-21 states, in part:

> It is the opinion of the Committee that when it is impractical to operate a pump at a specified reference point and adjust the resistance of the system to a specified reference point for flow rate, differential pressure or discharge pressure, the pump may be operated as close as practical to the specified reference point with the following requirements: The Owner shall adjust the system resistance to as close as practical to the specified reference point where the variance from the reference point does not exceed +2% or -1 % of the reference point when the reference point is flow rate, or +1% or -2% of the reference point when the reference point is differential pressure or discharge pressure.

Proposed Alternative Testing

The licensee seeks to perform future inservice pump testing in a manner consistent with the requirements as stated in ASME 0M Code Case OMN-21. Specifically, testing of all pumps identified in Table 1 above will be performed such that the flow rate is adjusted as close as practical to the reference value and within proceduralized limits of +2 percent or -1 percent of the reference flow rate, or alternatively, the differential pressure or discharge pressure is adjusted as close as practical to the reference value and within proceduralized limits of +1 percent or -2 percent of the reference discharge pressure or differential pressure.

The FitzPatrick plant operators will continue to strive to achieve the exact test reference values (flow, differential pressure, or discharge pressure) during testing. Typical test guidance will be to adjust the reference parameter (i.e., flow, differential pressure, or discharge pressure) to the specific reference value with additional guidance that if the reference value cannot be achieved with reasonable effort, the test will be considered valid if the steady-state flow rate is within the proceduralized limits of +2 percent or -1 percent of the reference value, or the steady-state discharge pressure or differential pressure is within the proceduralized limits of +1 percent or -2 percent of the reference value.

NRC Staff Evaluation

The NRC staff reviewed the information in Relief Request PRR-04, Revision O. The ASME 0M Code Case OMN-21 was developed to provide guidance on alternatives when it is impractical to operate a pump at a specified reference point for either flow rate, differential pressure, or discharge pressure. The pump may be operated as close as practical to the specified reference point with the following requirements.

Code Case OMN-21 was approved by the ASME Operation and Maintenance Standards Committee on April 20, 2012, with the NRC representative voting in the affirmative. The licensee proposed to adopt ASME 0M Code Case OMN-21. The applicability of Code Case OMN-21 is the ASME 0M Code, 1995 Edition through the 201 1 Addenda. The NRC staff notes that the language from ASME 0M Code Case OMN-21 has subsequently been included in the ASME 0M code, 2012 Edition, 2015 Edition, and 2017 Edition.

Based on its review, the NRC staff notes that in certain situations, it is not possible to operate a pump at a precise reference point. The NRC staff has reviewed the alternatives proposed in ASME 0M Code Case OMN-21 and found that the proposed alternatives are reasonable and appropriate when a pump cannot be operated at a specified reference point. Operation within the tolerance bands specified in ASME 0M Code Case OMN-21 provides reasonable assurance that licensees will be able to utilize the data collected to detect degradation of the pumps. Based on the NRC staff's review of ASME 0M Code Case OMN-21, and the licensee's commitment to use the bands specified in ASME 0M Code Case OMN-21 for flow rate, the NRC staff concludes that implementation of the alternatives contained in ASME 0M Code Case OMN-21 is acceptable for the pumps listed above. Therefore, the NRC staff concludes that the licensee's proposed alternative provides an acceptable level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that for Relief Requests PRR-01 and PRR-04, the proposed alternatives provide an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) for Relief Requests PRR-01 and PRR-04. Therefore, the NRC staff authorizes the use of the alternative requests for FitzPatrick for the fifth 10-year IST program interval, which is scheduled to begin on June 1, 2018, and end on September 30, 2027.

All other ASME 0M Code requirements for which relief was not specifically requested and approved in the subject requests for relief remain applicable.

Principal Contributor: Gurjendra Bedi

Date: April 13, 2018.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUESTS PRR-02, REVISION 0, AND PRR-03, REVISION 0,

FIFTH 10-YEAR INTERVAL INSERVICE TESTING PROGRAM

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated August 3, 2017, as supplemented by letter dated October 26, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML 17219A123 and ML 17299A560, respectively), Exelon Generation Company, LLC (Exelon or the licensee) submitted Relief Requests PRR-02, Revision 0, and PRR-03, Revision 0, to the U.S. Nuclear Regulatory Commission (NRC or the Commission). The licensee requested alternative tests in lieu of certain inservice testing (IST) requirements of the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (0M Code) for the IST program at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick) during the fifth 10-year IST program interval, which is scheduled to begin on June 1, 2018, and end on September 30, 2027.

Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative in Relief Request PRR-02, Revision 0, on the basis that the alternative provides an acceptable level of quality and safety. Pursuant to 10 CFR 50.55a(z)(2), the licensee requested to use the proposed alternative in Relief Request PRR-03, Revision 0, on the basis that the ASME 0M Code requirements present an undue hardship, without a compensating increase in the level of quality or safety.

2.0 REGULATORY EVALUATION

Section 50.55a(f), "Preservice and Inservice testing requirements," of 10 CFR requires, in part, that IST of certain ASME Code Class 1, 2, and 3 components must meet the requirements of the ASME 0M Code and applicable addenda, except where alternatives have been authorized pursuant to paragraphs 10 CFR50.55a(z)(1) or 10 CFR50.55a(z)(2).

Section 50.55a(z) of 10 CFR requires that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a, or portions thereof, must be submitted and authorized by the NRC prior to implementation. The applicant or licensee must demonstrate that: (1) the proposed alternative would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety. Section 50.55a of 10 CFR allows the NRC to authorize alternatives and to grant relief from ASME Code requirements upon making the necessary findings.

The guidance that the NRC staff considered in its review is NUREG/CP-0152, "Proceedings of the Fourth NRC/ASME Symposium on Valve and Pump Testing," dated July 15-18, 1996 (ADAMS Accession No. ML 18057B547), which provides acceptable guidance to licensees to address safety and technical issues related to valves and pumps.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternatives requested by the licensee.

3.0 TECHNICAL EVALUATION

The applicable ASME 0M Code edition and addenda for FitzPatrick during the fifth 10-year IST program interval is the 2004 Edition through the 2006 Addenda.

3.1 Licensee's Relief Request PRR-02, Revision 0

Applicable Code Requirements

Subsection ISTB-3300, "Reference Values," (a) states, "Initial reference values shall be determined from the results of testing meeting the requirements of IST B-3100, Preservice Testing, or from the results of the first inservice test."

Components for Which Relief is Requested

Table 1

Pump ID	Pump Description	ASME Code Class	ASME 0M Pump Group
10P-1A	Residual Heat Removal Service Water (RHRSW) Pump	3	A
10P-1B	RHRSW Pump	3	A
10P-1C	RHRSW Pump	3	A
10P-1D	RHRSW Pump	3	A

Reason for Alternative Request

The smooth running pumps listed in Table 1 have at least one vibration reference value (Vr) that is currently less than 0.05 inches per second (ips). A small value for Vr produces a small acceptable range for pump operation. The ASME 0M Code acceptable range limit for pump vibrations from Table ISTB-5221-1, "Vertical Line Shaft Centrifugal Pumps Test Acceptance Criteria," for both the Group A test and Comprehensive test is 2.5 Vr. Based on a small acceptable range, a smooth running pump could be subject to unnecessary corrective action if it exceeds this limit.

Subsection ISTB-6200, "Corrective Action," (a), "Alert Range," states:

> If the measured test parameter values fall within the alert range of Table ISTB-5121-1, Table ISTB-5221-1 , Table ISTB-5321-1 , or Table ISTB-5321-2, as applicable, the frequency of testing specified in IST B-3400 shall be doubled until the cause of the deviation is determined and the condition is corrected.

For very small vibration reference values, a significant portion of the vibration reading can be from flow variations, hydraulic noise, and instrument error, which can affect the repeatability of subsequent measurements. Also, experience gathered by the licensee's Predictive Maintenance (PdM) group has shown that changes in vibration levels in the range of 0.05 ips do not normally indicate significant degradation in pump performance.

Proposed Alternative Testing

The licensee seeks to apply a minimum value for Vr of 0.05 ips for the particular vibration measurement location. This minimum value would be applied to individual vibration locations for the residual heat removal service water (RHRSW) pumps listed in Table 1. The subsequent test results for that location will be compared to an alert range limit of 0.125 ips and a required action limit of 0.300 ips. These ranges, resulting from the proposed Vr of 0.05 ips and using the existing ASME 0M Code multipliers, shall be applied to vibration test results during both Group A tests and Comprehensive tests.

In addition to the requirements of the ASME 0M Code subsection IST B for IST, the pumps in the FitzPatrick IST program are also included in the FitzPatrick PdM program. The FitzPatrick PdM program currently uses predictive monitoring techniques such as vibration monitoring and analysis beyond that required by subsection ISTB bearing temperature trending, oil sampling and analysis, and/or thermography and analysis, as applicable.

If the measured parameters are outside the normal operating range or are determined by analysis to be trending toward an unacceptable degraded state, appropriate actions are taken that may include the following: initiate an issue report, increase monitoring to establish a rate of change, review component-specific information to identify the cause, and remove the affected pump from service to perform maintenance.

The licensee stated that the pumps in the IST program will remain in the FitzPatrick PdM program, even if certain pumps have very low vibration readings and are considered to be smooth running pumps.

NRC Staff Evaluation

The NRC staff reviewed the information in Relief Request PRR-02, Revision 0. The ASME 0M Code requires that the vibration of all pumps in a plant's IST program be measured. For centrifugal pumps, the measurements of each pump are taken in a plane approximately perpendicular to the rotating shaft in two orthogonal directions on each accessible pump-bearing housing. For vertical line shaft pumps, the vibration measurements are taken on the upper motor-bearing housing in three orthogonal directions, including the axial direction. The measurement is also taken in the axial direction on each accessible pump thrust-bearing housing. These measurements are to be compared with the ASME 0M Code vibration acceptance criteria to determine if the measured values are acceptable.

Subsection ISTB requires that, if during an inservice test, a bearing vibration measurement exceeds 2.5 Vr, the pump is considered in the alert range. The frequency of testing is then doubled until the condition is corrected and the vibration level returns below the alert range. Pumps whose vibration is recorded to be 6 Vr are considered in the required action range and must be declared inoperable until the cause of the deviation has been determined and the condition corrected. The vibration reference values are required to be determined when the pump is in good operating condition.

For pumps whose absolute magnitude of vibration is an order of magnitude below the absolute vibration limits established in subsection ISTB, a relatively small increase in vibration magnitude may cause the pump to enter the alert or required action range. These instances may be attributed to variation in flow, instrument accuracy, or other noise sources that would not be associated with degradation of the pump. Pumps that operate in this region are typically referred to as smooth running pumps. Based on a small acceptable range, a smooth running pump could be subject to unnecessary corrective action and additional testing.

Based on its review, the NRC staff finds that the alert and required action limits specified in the alternative request sufficiently address the previously undetected acute pump problems, and the licensee's PdM program appears to be designed to detect problems involving the mechanical condition, even well in advance of when the pump reaches its overall vibration alert limit.

Based on the experience gathered by the FitzPatrick PdM group, the licensee has proposed to establish a reference value of 0.05 ips. The use of the suggested reference value of 0.05 ips will provide an alert range of 0.125 to 0.30 ips, and the licensee's PdM program has shown that changes in vibration levels below 0.05 ips do not normally indicate significant degradation in pump performance. The reference value of 0.05 ips is consistent with previous NRC staff safety evaluations of similar issues. This alternative request js not for relief from the requirement to establish reference values, but from the method of determining the reference value. Therefore, the NRC staff concludes that the licensee's proposed alternative will provide an acceptable level of quality and safety.

3.2 Licensee's Alternative Request PRR-03, Revision 0

Applicable Code Requirements

Table IST B-5121-1, "Centrifugal Pump Test Acceptance Criteria," provides the vibration alert range low-end absolute limit of 0.325 ips for the Group A and Comprehensive tests.

Components for Which Relief is Requested

Table 2

Pump ID	Pump Description	ASME Code Class	ASME OM Pump Group
10P-3A	Residual Heat Removal (RHR) Pump	2	A
10P-3B	RHR Pump	2	A
10P-3C	RHR Pump	2	A
10P-3D	RHR Pump	2	A

Reason for Alternative Request

The increased periodicity of testing resulting from the 0.325 ips ASME 0M Code requirement is also an increase to the licensee's operations staff, plant scheduling, and adds run time to all RHR pumps. This request is based on analysis of vibration and pump differential pressure data indicating that no pump degradation is taking place.

Proposed Alternative Testing

The licensee seeks to apply an alternative vibration alert range low-end absolute limit of 0.408 ips for the RHR pumps listed in Table 2. The required action level for vibration will remain unchanged (> 6 V_r). The RHR pumps listed in Table 2 are tested using a full-flow recirculation test line back to the suppression pool for each Group A test and comprehensive pump tests. Based on the full-flow test line configuration, this test methodology results in flow-induced, broadband vibration readings greater than 0.325 ips, but less than the required action limits.

The guidance in NUREG/CP-OI 52 presented four key components that should be addressed in an alternative request of this type to streamline the review process. These four key components are as follows:

I. The licensee should have sufficient vibration history from the inservice testing, which verifies that the pump has operated at the vibration level for a significant amount of time, with any "spikes" in the data justified.

II. The licensee should have consulted with the pump manufacturer or vibration expert about the level of vibration the pump is experiencing to determine if the pump operation is acceptable.

III. The licensee should describe attempts to lower the vibration below the defined ASME 0M Code absolute levels through modifications to the pump.

IV. The licensee should perform a spectral analysis of the pump driver system to identify all contributors to the vibration levels.

The licensee provided a discussion of how it addressed these four key components in its submittal. Expert analyses and maintenance reviews have shown that this vibration has not resulted in degradation to the pump or motor. Data trends show that overall vibrations have remained steady since 1998.

The new Alert criterion value allows an alternative measure that still meets the intended function of monitoring the pump for degradation, while leaving the action levels as mandated by the ASME 0M Code. The proposed criterion encompasses the previous values that exceeded the Alert level, which would eliminate the unnecessary actions associated with exceeding the ASME 0M Code Alert limit when the pump is not seen as degrading. Any corrective actions triggered by vibrations between 0.408 ips to 0.7 ips will result in the same ASME 0M Code actions as previously required when exceeding the Alert limit of 0.325 ips.

NRC Staff Evaluation

The NRC staff reviewed the information in Relief Request PRR-03, Revision 0. Subsection ISTB-6200, "Corrective Action," (a), "Alert Range," requires that if the measured test parameter (in this case vibration) values fall within the alert range (greater than 0.325 ips through 0.7 ips) of ASME 0M Code Table ISTB-5121-1 , the frequency of testing specified in ISTB-3400 shall be doubled until the cause of the deviation is determined and the condition is corrected.

To accept pump vibration at a higher level than the ASME 0M Code-required alert range absolute limits, NUREG/CP-0152 recommends evaluating four key elements: (1) vibration history to verify that pumps were operated at this level of vibration for a significant amount of time with justification of "spikes" in test data; (2) consulting with the pump manufacturer/vibration experts to verify that the vibration levels of the pumps are acceptable; (3) attempts to lower the vibration level through modifications to the pumps or the system and structures of the pumps; and (4) perform spectral analysis to identify all contributors to the vibration level. In its submittal, the licensee provided information to address each of these key elements. The licensee also included its evaluation of all of these four key elements for the RHR pumps.

The licensee stated that the pump vendor was contacted during the initial investigation of the cause for failed vibration acceptance criteria. The licensee also stated that the basis for the 0.408 inch/second (in/sec) alert limit comes from the Technical Associates of Charlotte recommendations for vertical pumps. The RHR pump vendor did not recommend a specific value regarding the increased vibration alert limit but stated that the pumps should not be adversely impacted, provided that no upward trend existed in the vibration measurement data.

Also, the data provided in the alternative request in PRR-03, Revision O, shows that the ASME 0M Code alert range value of 0.325 in/sec has been exceeded only on pumps 1 OP-3B and 1 OP-3C. When this same alternative request was submitted as PRR-05 for the fourth 10-year IST interval on February 21 , 2014 (ADAMS Accession No. ML 14057A553), the NRC staff asked the licensee to provide justification on why the alternative request is necessary for RHR pumps 1 OP-3A and 10P-3D. In its response dated July 31, 2014 (ADAMS Accession No. ML 14213A115), the licensee stated that RHR pumps 10P-3A and 10P-3C are common to Train A, and RHR pumps 10P-3B and 10P-3D are common to Train B. The licensee further stated that the FitzPatrick IST program implementing procedures require increased frequency testing of both pumps in each particular train if the ASME 0M Code alert range value is exceeded on one of the pumps.

Based on its review, the NRC staff found that the licensee has submitted sufficient vibration history to verify that the pumps have operated at this vibration level for a significant period of time with no adverse impacts on performance. Spike data has been justified by consultation with an independent pump expert. The licensee has described attempts to reduce vibration and has demonstrated that the cause of the vibration appears to be the vane pass frequency inherent to the pump design. Spectral analysis of the pump-driver system was performed to identify all contributors to vibration levels. Based on the evaluation of the provided historical pump vibration data, the NRC staff concluded that these are not indicative of degraded pump performance.

The licensee has proposed to raise the minimum vibration alert range for the four RHR pumps listed in Table 2 from 0.325 ips to 0.408 ips. The NRC staff reviewed the historical vibration information for the four RHR pumps and noted that the vibration parameters cited in the alternative request for RHR pumps 10-P3B and 10-P3C do occasionally exceed the ASME 0M Code, Table ISTB-5121-1 minimum alert level of 0.325 ips alert limit. The analysis and evaluation that the licensee performed provide reasonable assurance of operational readiness. Additionally, the proposed alternative alert limit of 0.408 ips is below the required action limit of

0.700 ips, and the licensee has demonstrated that these pumps have a normal operational history at this vibration level with no adverse consequences.

Based on the NRC staffs review of the historical vibration data provided, the additional PdM activities proposed, and the identification of vane pass frequency as a primary contributor to vibration, the NRC staff finds that implementation of the proposed alternative is acceptable for the RHR pumps listed in Table 2. Therefore, the NRC staff concludes that compliance with the specified ASME 0M Code requirement would result in hardship without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determined that for Relief Request PRR-02, Revision 0, the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) for Relief Request PRR-02, Revision O. Therefore, the NRC staff authorizes the use of the alternative request for FitzPatrick for the fifth 10-year IST program interval, which is scheduled to begin on June 1, 2018, and end on September 30, 2027.

In addition, as set forth above, the NRC staff determined that for Relief Request PRR-03, Revision 0, the proposed alternative provides reasonable assurance that the affected components are operationally ready. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2) for Relief Request PRR-03, Revision 0. Therefore, the NRC staff authorizes the use of the alternative request for FitzPatrick for the fifth 10-year IST program interval, which is scheduled to begin on June 1, 2018, and end on September 30, 2027.

All other ASME 0M Code requirements for which relief was not specifically requested and approved in the subject requests remain applicable.

Principal Contributor: Robert Wolfgang

Date: April 13, 2018.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVE REQUEST VRR-03, REVISION 0

FIFTH 10-YEAR INSERVICE TESTING INTERVAL

EXELON GENERATION COMPANY, LLC

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1 .0 INTRODUCTION

By letter dated August 3, 2017, as supplemented by letter dated October 26, 2017 (Agencywide Documents Access and Management System Accession Nos. ML 17219A123 and ML 17299A560, respectively), Exelon Generation Company, LLC (Exelon or the licensee) submitted Relief Request VRR-03, Revision O, to the U.S. Nuclear Regulatory Commission (NRC or the Commission). The licensee requested alternative tests in lieu of certain inservice testing (IST) requirements of the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (0M Code) for the IST program at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick) during the fifth 10-year IST program interval, which is scheduled to begin on June 1, 2018, and end on September 30, 2027. Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternatives in Relief Request VRR-03, Revision O, on the basis that the alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Section 50.55a(f), "Preservice and Inservice testing requirements," of 10 CFR requires, in part, that IST of certain ASME Code Class 1, 2, and 3 components must meet the requirements of the ASME 0M Code and applicable addenda, except where alternatives have been authorized pursuant to paragraphs 10 CFR 50.55a(z)(1) or 10 CFR 50.55a(z)(2).

Section 50.55a(z) of 10 CFR states that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a, or portions thereof, must be submitted and authorized by the NRC prior to implementation. The applicant or licensee must demonstrate that: (1) the proposed alternative would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety. Section 50.55a of 10 CFR allows the NRC to authorize alternatives and to grant relief from ASME Code requirements upon making the necessary findings.

Enclosure 3

The guidance that the NRC staff considered in its review is Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME 0M Code," June 2003 (ADAMS Accession No. ML030730430), which provides approved for use as voluntary alternatives to the mandatory ASME 0M Code provisions that are incorporated by reference 10 CFR Part 50.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

The applicable ASME 0M Code Edition and Addenda for FitzPatrick during the fifth 10-year IST program interval is the 2004 Edition through the 2006 Addenda.

3.1 Licensee's Relief Request VRR-03, Revision O

Applicable Code Requirements

Mandatory Appendix I, paragraph I-1 320, "Test Frequencies, Class 1 Pressure Relief Valves," states, in part, that "Class 1 pressure relief valves shall be tested at least once every 5 years, starting with initial electric power generation."

Components for Which Relief is Requested

Class 1, Target Rock pilot-operated main steam safety/relief valves (SRVs):

02RV-071A	02RV-071B	02RV-071C	02RV-071D
02RV-071E	02RV-071 F	02RV-071G	02RV-071H
02RV-071J	02RV-071K	02RV-071L	

Reason for Alternative Request

The current 24-month operating cycle would require the removal and test of 50 percent of the SRVs every refueling outage (i.e., five or six of 1 1), such that all valves are removed and tested every two refueling outages. Approval of extending the test interval for the valves to 6 years with a grace period of 6 months, consistent with ASME 0M Code Case OMN-17, "Alternative Rules for Testing ASME Class 1 Pressure Relief/Safety Valves," would reduce the minimum number of SRVs tested at FitzPatrick over three refueling outages.

Proposed Alternative Testing

As an alternative to the ASME 0M Code required 5-year test interval, the licensee proposed that Class 1 pressure relief valves, 02RV-071A, B, C, D, E, F, G, H, J, K, and L, be tested at least once every three refueling cycles with a minimum of 20 percent of the valves tested within any 24-month interval. This 20 percent would consist of valves (complete assemblies) that have not been tested during the current 72-month interval, if they exist. The test interval for any individual valve would not exceed 72 months except that a 6-month grace period is allowed to coincide with refueling outages to accommodate extended shutdown periods. The licensee proposed to continue testing all 11 installed pilot valves every refueling outage.

The relief valve testing and maintenance cycle at Fitzpatrick consists of an as-found inspection, seat leakage, and set pressure testing. After as-found set pressure testing, the valves shall be disassembled and inspected to verify that parts are free of defects resulting from time-related degradation or service-induced wear. As-left set pressure testing shall be performed following maintenance and prior to returning the valve to service.

3.2 NRC Staff Evaluation

The NRC staff reviewed the information in Relief Request VRR-03, Revision 0. The FitzPatrick SRVs are ASME Code Class 1 pressure relief valves that provide overpressure protection for the reactor coolant pressure boundary to prevent unacceptable radioactive release and exposure to plant personnel. Mandatory Appendix I of the ASME 0M Code requires that Class 1 pressure relief valves be tested at least once every 5 years.

However, Mandatory Appendix I does not require that pressure relief valves be disassembled and inspected prior to the start of the 5-year test interval. In lieu of the 5-year test interval, the licensee proposed to implement ASME 0M Code Case OMN-17, which allows a test interval of 6 years plus a 6-month grace period. The ASME Committee on 0M developed Code Case OMN-17 and published it in the 2009 Edition of the ASME 0M Code. ASME 0M Code Case OMN-17 imposes a special maintenance requirement to disassemble and inspect each pressure relief/safety valve to verify that parts are free from defects resulting from time-related degradation or service-induced wear prior to the start of the extended test interval and at each required test during the interval.

Code Case OMN-17 has not yet been added to Regulatory Guide 1.192 or included in 10 CFR 50.55a by reference. However, the NRC has allowed licensees to use ASME OM Code Case OMN-17, provided all requirements in the code case are met. This maintenance will also help to reduce the potential for setpoint drift and increase the reliability of these SRVs to perform their design requirement functions.

Furthermore, ASME 0M Code Case OMN-17 is performance-based in that it requires that the SRVs be tested more frequently if test failures occur. For example, ASME 0M Code Case OMN-17 requires that two additional valves be tested when a valve in the initial test group exceeds the set pressure acceptance criteria. All remaining valves in the group are required to be tested if one of the additional valves tested exceeds its set pressure acceptance criteria.

Based on its review, the NRC staff finds that implementation of ASME 0M Code Case OMN-17 for the testing of the FitzPatrick SRVs, in lieu of the requirements of the 2004 Edition through the 2006 Addenda, Mandatory Appendix I, paragraph 1-1320, of the ASME 0M Code, provides an acceptable level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that for Relief Request VRR-03, Revision 0, the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) for Relief Request VRR-03, Revision 0. Therefore, the NRC staff authorizes the use of the alternative request for FitzPatrick for the fifth 10-year IST program interval, which is scheduled to begin on June 1, 2018, and end on September 30, 2027.

All other ASME 0M Code requirements for which relief was not specifically requested and approved in the subject request for relief remain applicable.

Principal Contributor: John Billerbeck

Date: April 13, 2018.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUESTS VRR-01 REVISION VRR-02 REVISION 0; AND VRR-04, REVISION 0

FIFTH 10-YEAR INTERVAL INSERVICE TESTING PROGRAM

EXELON GENERATION COMPANY, LLC

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1 .0 INTRODUCTION

By letter dated August 3, 2017, as supplemented by letter dated October 26, 2017 (Agencywide Documents Access and Management System Accession Nos. ML 17219A123 and ML 17299A560, respectively), Exelon Generation Company, LLC (Exelon or the licensee) submitted Relief Requests VRR-01, Revision O; VRR-02, Revision O; and VRR-04, Revision O, to the U.S. Nuclear Regulatory Commission (NRC or the Commission). The licensee requested alternative tests in lieu of certain inservice testing (IST) requirements of the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (0M Code) for the IST program at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick) during the fifth 10-year IST program interval, which is scheduled to begin on June 1, 2018, and end on September 30, 2027.

Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(z)(1), the licensee requested to use proposed alternatives in Relief Request VRR-02, Revision 0, and Relief Request VRR-04, Revision 0, on the basis that the alternatives provide an acceptable level of quality and safety. Pursuant to 10 CFR 50.55a(z)(2), the licensee requested to use the proposed alternative in Relief Request VRR-01 , Revision 0, on the basis that the ASME 0M Code requirements present an undue hardship, without a compensating increase in the level of quality or safety.

2.0 REGULATORY EVALUATION

Section 50.55a(f), "Preservice and Inservice testing requirements," of 10 CFR requires, in part, that IST of certain ASME Code Class 1, 2, and 3 components must meet the requirements of the ASME 0M Code and applicable addenda, except where alternatives have been authorized pursuant to paragraphs 10 CFR 50.55a(z)(1) or 10 CFR 50.55a(z)(2).

Section 50.55a(z) of 10 CFR states that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a, or portions thereof, must be submitted and authorized by the NRC prior to implementation. The applicant or licensee must demonstrate that: (1) the proposed alternative would provide an acceptable level of quality and safety, or (2) compliance

Enclosure 4

with the specified requirements would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety. Section 50.55a of 10 CFR allows the NRC to authorize alternatives and to grant relief from ASME Code requirements upon making the necessary findings.

The guidance that the NRC staff considered in its review include the following:

- NUREG-0933, "Resolution of Generic Safety Issues," Issue 105, "Interfacing Systems LOCA [Loss-of-Coolant Accident] at LWRs [Light-Water Reactors]," December 201 1 (ADAMS Accession No. ML 1 1353A382), which provides a single-source repository of all NRC generic safety issue reviews.

- NUREG-1482, Revision 2, "Guidelines for Inservice Testing at Nuclear Power Plants," October 2013 (ADAMS Accession No. ML 13295A020), which provides acceptable guidance to licensees to establish a basic understanding of the regulatory basis for pump and valve IST programs.

- NUREG/CR-5928, "Final Report of the ISLOCA Research Program," August 2007 (ADAMS Accession No. Ml-072430731), which quantifies the risk associated with an interfacing system loss-of-coolant accident (ISLOCA) event.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternatives requested by the licensee.

3.0 TECHNICAL EVALUATION

The applicable ASME 0M Code Edition and Addenda for FitzPatrick during the fifth 10-year IST program interval is the 2004 Edition through the 2006 Addenda.

3.1 Licensee's Relief Request VRR-01, Revision O

Applicable Code Requirements

Subsection ISTC-5151 , "Valve Stroke Testing," (a), states that "Active valves shall have their stroke times measured when exercised in accordance with ISTC-3500."

Subsection ISTC-5151 , "Valve Stroke Testing," (c), states that "Stroke time shall be measured to at least the nearest second."

Components for Which Relief is Requested

Table 1

Valve ID	Function	Category	Class
07SOV-104A	Traversing In-Core Probe (TIP) Containment Isolation Valve (CIV)	A	2
07SOV-104B	TIP CIV	A	2
07sov-104C	TIP CIV	A	2

Reason for Alternative Request

The Category A containment isolation solenoid operated valves identified in this request have no safety function in the open direction as they open to allow the passage of the TIP assembly and drive cable for flux mapping operations. These valves have an active safety function in the closed direction in response to a primary containment isolation system signal to seal the TIP guide tubes. Therefore, an exercise test and subsequent stroke time test are only required in the closed direction.

The design of the TIP control system does not allow for measurement of the closure stroke times of valves 07SOV-104A, B, and C. Measuring the closure stroke times in accordance with the ASME 0M Code would require a costly computer control system modification. Closure of valves 07SOV-104A, B, and C could also be accomplished by an alternative method, but this method would require manual extraction and retraction of the TIP from the shield block. This method of testing would be contrary to the principles of keeping radiation exposure as low as reasonably achievable because it would result in radiation exposure to personnel performing the test.

The proposed alternative test ensures the operation of valves 07SOV-104A, B, and C in both directions and provides an acceptable level of quality and safety. This method meets the desired outcome of monitoring valve stroke time for degradation since the computer controls the 10-second delay and the additional approximate 2 seconds for valve closure should indicate the actual stroke time.

Proposed Alternative Testing

As an alternative to the ASME 0M Code required 5-year test interval, the licensee proposed to measure overall cycle time (opened and closed) for the TIP CIVs 07SOV104A, B, and C, in accordance with Subsection ISTC-5152. Exelon will time the opening (10-second delay time included) and closing cycle for valves 07SOV-104A, B, and C. The time from open initiation to receipt of the closed light for each valve will be monitored with a stop watch.

NRC Staff Evaluation

The NRC staff reviewed the information in Relief Request VRR-01 , Revision 0. Subsection ISTC-5151 of ASME 0M Code details the requirements for valve stroke testing of solenoid operated valves. In lieu of these requirements, the licensee proposed to full stroke exercise each valve noted in Table 1 and determine proper operation by using the computer control system for the TIP solenoid valves, which includes a provision for measuring valve cycle time (opened and closed). The computer control system opens the subject valve, maintains it energized for 10 seconds, and deenergizes the valve allowing the solenoid valve to stroke closed (< 2 seconds). The total computer control system test is 12 seconds. Exelon will apply Subsection ISTC-5152(a), which requires that each valve exhibit no more than ± 25 percent change in stroke time when compared to the reference value, except that the full-stroke limiting time for each valve will be truncated at 12 seconds.

Based on its review, the NRC staff finds that the proposed alternative is consistent with the guidance in NUREG-1482, Revision 2, paragraph 4.2.3. Therefore, the NRC staff concludes that the proposed alternative provides an acceptable level of quality and safety.

3.2. Licensee's Relief Request VRR-02, Revision 0

Applicable Code Requirements

Subsection ISTC-3510, "Exercising Test Frequency," states, in part, that "Active Category A, Category B, and Category C check valves shall be exercised nominally every 3 months, except as provided by ISTC-3520, ISTC-3540, ISTC-3550, ISTC-3570, ISTC-5221 , and ISTC-5222."

Subsection ISTC-3522, "Category C Check Valves," (c), states that "If exercising is not practicable during operation at power and cold shutdowns, it shall be performed during refueling outages."

Subsection ISTC-3700, "Position Verification Testing," states, in part, that "Valves with remote position indicators shall be observed locally at least once every 2 years to verify that valve operation is accurately indicated. "

Components for Which Relief is Requested

Table 2

Valve ID	System	Category	Class
02-2EFV-PS-128A	Reactor Water Recirculation (RWR) Excess Flow Check Valve (EFCV)	A/C	1
02-2EFV-PS-128B	RWR EFCV	A/C	1
02-2EFV-PT-24A	RWR EFCV	A/C	1
02-2EFV-PT-24B	RWR EFCV	A/C	1
02-2EFV-PT-25A	RWR EFCV	A/C	1
02-2EFV-PT-25B	RWR EFCV	A/C	1
02-2EFV1-DPT-11 IA	RWR EFCV	A/C	1
02-2EFV1-DPT-111 B	RWR EFCV	NC	1
02-2EFV1_FT-I I OA	RWR EFCV	A/C	1
02-2EFV1-FT-1 IOC	RWR EFCV	A/C	1
02-2EFV1_FT-11 OE	RWR EFCV	A/C	1
02-2EFV1-FT-11 OG	RWR EFCV	A/C	1
02-2EFV2-DPT-11 IA	RWR EFCV	A/C	1
02-2EFV2-DPT-111 B	RWR EFCV	A/C	1
02-2EFV2-FT-11 OA	RWR EFCV	A/C	1
02-2EFV2-FT-11 oc	RWR EFCV	A/C	1
02-2EFV2-FT-11 OE	RWR EFCV	A/C	1
02-2EFV2-FT-11 OG	RWR EFCV	A/C	1
02-3EFV-11	Nuclear Boiler (NB) EFCV	A/C	1
02-3EFV-13A	NB EFCV	A/C	1
02-3EFV-13B	NB EFCV	A/C	1
02-3EFV-15A	NB EFCV	A/C	1
02-3EFV-15B	NB EFCV	A/C	1

Valve ID	System	Category	Class
02-3EFV-15N	NB EFCV	A/C	1
02-3EFV-17A	NB EFCV	A/C	1
02-3EFV-17B	NB EFCV	A/C	1
02-3EFV-19A	NB EFCV	A/C	1
02-3EFV-19B	NB EFCV	A/C	1
02-3EFV-21A	NB EFCV	A/C	1
02-3EFV-21B	NB EFCV	A/C	1
02-3EFV-21C	NB EFCV	A/C	1
02-3EFV-21D	NB EFCV	A/C	1
02-3EFV-23A	NB EFCV	A/C	1
02-3EFV-23B	NB EFCV	A/C	1
02-3EFV-23C	NB EFCV	A/C	1
02-3EFV-23D	NB EFCV	A/C	1
02_3EFV-23	NB EFCV	A/C	1
02-3EFV-25	NB EFCV	A/C	1
02-3EFV-31A	NB EFCV	A/C	1
02-3EFV-31B	NB EFCV	A/C	1
02-3EFV-31C	NB EFCV	A/C	1
02-3EFV-31D	NB EFCV	A/C	1
02-3EFV-31E	NB EFCV	A/C	1
02-3EFV-31 F	NB EFCV	A/C	1
02-3EFV-31G	NB EFCV	A/C	1
02-3EFV-31 H	NB EFCV	A/C	1
02-3EFV-31J	NB EFCV	A/C	1
02-3EFV-31K	NB EFCV	A/C	1
02-3EFV-31L	NB EFCV	A/C	1
02_3EFV-31 M	NB EFCV	A/C	1
02-3EFV-31N	NB EFCV	A/C	1
02-3EFV-31P	NB EFCV	A/C	1
02-3EFV-31R	NB EFCV	A/C	1
02-3EFV-31S	NB EFCV	A/C	1
02-3EFV-33	NB EFCV	A/C	1
13EFV-01A	NB EFCV	A/C	1
13EFV-01B	NB EFCV	A/C	1
13EFV-02A	NB EFCV	A/C	1
13EFV-02B	NB EFCV	A/C	1
14EFV-31A	NB EFCV	A/C	1
14EFV-31B	NB EFCV	AC	1
23EFV-01A	High Pressure Coolant Injection (HPCI) EFCV	AC	1

Valve ID	System	Category	Class
23EFV-01B	HPCI EFCV	A/C	1
23EFV-02A	HPCI EFCV	A/C	1
23EFV-02B	HPCI EFCV	A/C	1
29EFV-30A	Main Steam (MS) EFCV	A/C	1
29EFV-30B	MS EFCV	A/C	1
29EFV-30C	MS EFCV	A/C	1
29EFV-30D	MS EFCV	A/C	1
29EFV-34A	MS EFCV	A/C	1
29EFV-34B	MS EFCV	A/C	.1
29EFV-34C	MS EFCV	A/C	1
29EFV-34D	MS EFCV	A/C	1
29EFV-53A	MS EFCV	A/C	1
29EFV-53B	MS EFCV	A/C	1
29EFV-53C	MS EFCV	A/C	1
29EFV-53D	MS EFCV	A/C	1
29EFV-54A	MS EFCV	A/C	1
29EFV-54B	MS EFCV	A/C	1
29EFV-54C	MS EFCV	A/C	1
29EFV-54D	MS EFCV	A/C	1

Reason for Alternative Request

The ASME 0M Code requires check valves to be exercised quarterly during plant operation, or if valve exercising is not practicable during plant operation and cold shutdown, it shall be performed during refueling outages. Based on past experience, EFCV testing during in-service leakage testing can become the outage critical path and could possibly extend the outage if all EFCVs were to be tested during this time frame. The testing requires isolation of the instruments associated with each EFCV and opening of a drain valve to actuate the EFCV. Process fluid will be contaminated to some degree, requiring special measures to collect flow from the drain valve and also contributes to an increase in personnel radiation exposure.

Proposed Alternative Testing

The licensee proposed to exercise test, by full-stroke to the position required to fulfill its function, a representative sample of EFCVs every refuel outage. The representative sample is based on approximately 20 percent of the valves each cycle such that each valve is tested at least once every 10 years (nominal). Industry experience as documented in General Electric (GE) Topical Report NEDO-32977-A/B21-00658-01 , "Excess Flow Check Valve Testing Relaxation, June 2000 (ADAMS Accession No. ML003729011), indicates that EFCVs have a very low failure rate.

The instrument lines at Fitzpatrick have a flow restricting orifice upstream of the EFCVs to limit reactor water leakage in the event of rupture. The Fitzpatrick Final Safety Analysis Report (FSAR), paragraph 7.1.6, "Supplemental NSSS Supplier Information," does not credit the EFCVs, but instead credits the installed orifice for limiting the release of reactor coolant following an instrument line break. Thus, a failure of an EFCV, though not expected as a result of this request, is bounded by the FSAR analysis. The licensee's test experience is consistent with the findings in NEDO-32977-A. The NEDO-32977-A topical report indicates similarly that many reported test failures at other plants were related to test methodologies and not actual EFCV failures.

The EFCV failures will be documented in the Fitzpatrick's Corrective Action Program as an equipment and surveillance test failure. The failure will be evaluated and corrected to ensure EFCV performance remains consistent with the extended test interval, a minimum acceptance criteria of less than or equal to one failure per year on a 3-year rolling average will be required.

NRC Staff Evaluation

The EFCVs are installed on instrument lines to limit the release of fluid in the event of an instrument line break. The EFCVs are not required to close in response to a containment isolation signal and are not required to operate under post-LOCA conditions. The EFCVs are required to be tested in accordance ASME 0M Code ISTC-3510. The proposed change revises the surveillance frequency by allowing a "representative sample" of EFCVs to be tested every refueling outage.

The NRC staff reviewed NEDO-32977-A and issued its safety evaluation on March 14, 2000 (ADAMS Accession No. Ml-003691722). In its safety evaluation, the NRC staff found that the test interval could be extended up to a maximum of 10 years. In conjunction with this finding, the NRC staff noted that each licensee that adopts the relaxed test interval program for EFCVs must have a failure feedback mechanism and corrective action program to ensure EFCV performance continues to be bounded by the topical report results. Also, each licensee is required to perform a plant-specific radiological dose assessment, EFCV failure analysis, and release frequency analysis to confirm that they are bounded by the generic analyses of the topical report.

The NRC staff reviewed the licensee's current proposal and a previous NRC-approved request for the fourth IST interval dated November 27, 2007 (ADAMS Accession No. Ml-072910422), for its applicability to GE Nuclear's topical report NEDO-32977-A, as well as conformance with the NRC staff's guidance regarding radiological dose assessment, EFCV failure rate, release frequency, and the proposed failure feedback mechanism and corrective action program.

Based on its review, the NRC staff concludes that the radiological consequences of an EFCV failure are sufficiently low and acceptable, and that the alternative testing in conjunction with the corrective action plan provides a high degree of valve reliability and operability. Therefore, the NRC staff finds that the licensee's proposed test alternative provides an acceptable level of quality and safety.

3.3. Licensee's Relief Request VRR-04, Revision O

Applicable Code Requirements

Subsection ISTC-3630, "Leakage Rate for Other Than Containment Isolation Valves," states, in part, that "Category A valves with a leakage requirement not based on an Owner's 10 CFR 50, Appendix J program, shall be tested to verify their seat leakages are within acceptable limits.

Subsection ISTC-3630(a), "Frequency," states that "Tests shall be conducted at least once every 2 years."

Components for Which Relief is Requested

Table 3

Valve ID	Function	Category	Class
10AOV-68A	Residual Heat Removal (RHR) A Low Pressure Coolant Injection (LPCI) Testable Check Valve	A/C	1
10AOV-68B	RHR A LPCI Testable Check Valve	A/C	1
10MOV-25A	RHR A LPCI Inboard Injection Valve	A	1
10MOV-25B	RHR B LPCI Inboard Injection Valve	A	1
14AOV-13A	CSP A Reactor Isolation Testable Check Valve	A/C	1
14AOV-13B	CSP B Reactor Isolation Testable Check Valve	A/C	1
14MOV-12A	Core Spray Loop A Inboard Isolation Valve	A	1
14MOV-12B	Core Spray Loop B Inboard Isolation Valve	A	1
10MOV-17	RHR Shutdown Cooling Outboard Isolation Valve	A	1
10MOV-18	RHR Shutdown Cooling Inboard Isolation Valve	A	1

Reason for Alternative Request

At FitzPatrick, pressure isolation valves (PIVs) are Category A or Category A/C valves within the scope of Subsection ISTC-3630 of the ASME 0M Code. This alternative to allow for scheduling of leak tests for the valves identified in Table 3 to a performance-based frequency that is the same as 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B, "Performance-Based Requirements," testing, would provide an acceptable level of quality and safety.

Proposed Alternative Testing

The licensee proposed an alternative test frequency in lieu of the requirements found in Subsection ISTC-3630(a) for the ten applicable PIVs listed in Table 3. Appendix J to 10 CFR Part 50 was amended to improve the focus of the body of regulations by eliminating prescriptive requirements that are marginal to safety and to provide licensees greater flexibility for cost-effective implementation methods for meeting regulatory safety objectives.

Nuclear Energy Institute (NEI) 94-01, Revision 3A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 2012 (ADAMS Accession No. ML 12221 A202), describes the risk-informed basis for extended test intervals under Option B. That justification documents valves that have demonstrated good leakage rate performance over two consecutive cycles are subject to future failures predominantly governed by the random failure rate of the component. The NEI 94-01, Revision 3A guidance also presents the results of a comprehensive risk analysis, including the statement that "The risk impact associated with increasing test intervals is negligible (less than 0.1 percent of total risk)."

The guidance in NUREG-0933 discussed the need for PIV leak-rate testing based primarily on three pre-1980 historical failures of applicable valves industrywide. These failures all involved human errors in either operations or maintenance. None of these failures involved inservice equipment degradation. The performance of PIV leak-rate testing provides assurance of acceptable seat leakage with the valve in a closed condition. Typical PIV leak-rate testing does not identify functional problems that may inhibit the valve's ability to reposition from open to close.

Fitzpatrick proposed to perform PIV testing at intervals specified in NEI 94-01. Program guidance will be established such that if any of the valves fail either the CIV test or PIV test, the test interval for both tests will be reduced to once every 30 months until they can be reclassified as good performers per the performance evaluation requirements of Appendix J, Option B. The test intervals for the valves identified in this request will be determined in the same manner as is done for CIV testing under Option B. The test interval may be extended upon completion of two consecutive periodic PIV tests with results within the prescribed acceptance criteria. Any PIV test failure will require a return to the initial interval until good performance can again be established.

The risks associated with extending the leakage test interval to a maximum of 75 months are extremely low. The basis for this alternative request is the historically good performance of the PIVs. This alternative will also provide significant reductions in radiation dose.

NRC Staff Evaluation

The NRC staff reviewed the information in Relief Request VRR-04, Revision O. The licensee proposed to functionally test and verify the leakage rate of these PIVs using the 10 CFR Part 50, Appendix J, Option B performance-based schedule. Valves would initially be tested at the required interval schedule, which is currently every refueling outage, or 2 years, as specified by the ASME 0M Code, Section ISTC-3630(a). Valves that have demonstrated good performance for two consecutive cycles may have their test interval extended to 75 months. Any PIV leakage test failure would require the component to return to the initial interval of every refueling outage, or 2 years, until good performance can again be established.

Pressure isolation valves are defined as two valves in series within the reactor coolant pressure boundary, which separate the high pressure reactor coolant system from an attached lower pressure system. Failure of a PIV could result in an over-pressurization event that could lead to a system rupture and possible release of fission products to the environment. This type of failure event was analyzed under the NUREG/CR-5928 ISLOCA research program. The NUREG/CR-5928 research program analyzed boiling-water reactor (BWR) and pressurized-water reactor designs. The conclusion of the analysis resulted in ISLOCA not being a risk concern for BWR design. FitzPatrick is a BWR design.

Appendix J, Option B to 10 CFR Part 50, is a performance-based leakage test program. Guidance for implementation of acceptable leakage rate test methods, procedures, and analyses is provided in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995 (ADAMS Accession No. Ml-003740058). Regulatory Guide 1.163 endorses NEI 94-01 , Revision O, "Industry Guideline for Implementing Performance Based Option of 10 CFR 50, Appendix J," dated July 21 , 1995 (ADAMS Accession No. ML 1 1327A025), with the limitation that Type C components test interval cannot extend greater than 60 months. The current version of NEI 94-01 is Revision 3-A, which allows Type C containment isolation valves test intervals to be extended to 75 months, with a permissible extension for nonroutine emergent conditions of 9 months (84 months total). The NRC staff finds the guidance in NEI 94-01, Revision 3-A, acceptable (safety evaluation dated June 8, 2012, and approval letter dated December 2, 2016; available at ADAMS Accession Nos. ML 121030286 and ML 12226A546, respectively), with the following conditions:

1. Extended interval for Type C local leakage-rate tests (LLRTs) may be increased to 75 months, with the requirement that a licensee's post-outage report include the margin between Type B and Type C leakage rate summation and its regulatory limit. In addition, a corrective action plan shall be developed to restore the margin to an acceptable level. Extensions of up to 9 months (total maximum interval of 84 months for Type C tests) are permissible only for nonroutine emergent conditions. This

provision (9-month extension) does not apply to valves that are restricted and/or limited to 30-month intervals in NEI 94-01 , Revision 3A, Section 10.2, "Type B and Type C Testing Frequencies" (such as BWR main steam isolation valves), or to valves held to the base interval (30 months) due to unsatisfactory LLRT performance.

2. When routinely scheduling any LLRT valve interval beyond 60 months and up to 75 months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B and Type C total and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

Based on its review, the NRC staff finds that the proposed alternative was previously authorized for use at FitzPatrick for the fourth IST program interval under safety evaluation dated March 16, 2012 (ADAMS Accession No. ML 12072A1 13). As noted in the licensee's proposed alternative, the valves have maintained a history of good performance. Extending the leakage test interval based on good performance and the low risk factor as noted in NUREG/CR-5928 is a logical progression to a performance-based program. Therefore, the NRC staff concludes that the licensee's proposed alternative provides an acceptable level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that for Relief Request VRR-02, Revision O, and Relief Request VRR-04, Revision O, the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) for Relief Request VRR-02, Revision O, and Relief Request VRR-04, Revision O. Therefore, the NRC staff authorizes the use of the alternative request for FitzPatrick for the fifth 10-year IST program interval, which is scheduled to begin on June 1, 2018, and end on September 30, 2027.

As set forth above, the NRC staff determines that for Relief Request VRR-01 , Revision O, the proposed alternative provides reasonable assurance that the affected components are operationally ready. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2) for Relief Request VRR-01, Revision O. Therefore, the NRC staff authorizes the use of the alternative request for FitzPatrick for the fifth 10-year IST program interval, which is scheduled to begin on June 1, 2018, and end on September 30, 2027.

All other ASME 0M Code requirements for which relief was not specifically requested and approved in the subject requests remain applicable.

Principal Contributor: Michael Farnan

Date: April 13, 2018

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - ISSUANCE OF RELIEF REQUESTS FOR ALTERNATIVES TO CERTAIN ASME 0M CODE REQUIREMENTS (CAC NOS. MG0052, MG0053, MG0054, MG0055, MG0056, MG0057, MG0058, AND MG0061; EPID L-2017-LLR-0067, L-2017-LLR-0068, L-2017-LLR-0069, L-2017-LLR-0070, L-2017-LLR-0071 , L-2017-LLR-0072, L-2017-LLR-0073, AND L-2017-LLR-0074) DATED APRIL 13, 2018

DISTRIBUTION:
PUBLIC RidsACRS MailCTR
RidsNrrDeEmib
RidsNrrDorlLp11
RidsNrrLAJBurkhardt RidsNrrLALRonewicz
RidsNrrPMFitzPatrick
RidsRgn1 MailCenter
RW01fgang, NRR
GBedi, NRR
JBowen, OEDO
LBurkhart, OEDO
MFarnan, NRR MGray, R-l

ADAMS Accession No.: ML18044A993 *b safet evaluation

OFFICE	NRR/DORL/LPLI/PM	NRR/DORL/LPLI/PM	NRR/DORL/LPLI/LA
NAME	BVenkataraman	THood	LRonewicz (JBurkhardt for)
DATE	02/13/2018	04/4/2018	04/04/2018
OFFICE	NRR/DE/EMIB/BC*	NRR/DORL/LPLI/BC	
NAME	SBailey	JDanna	
DATE	01/09/2018	04/13/2018	

OFFICIAL RECORD COPY

<u>ATTACHMENT 7</u>

<u>CODE CASE INDEX</u>

CODE CASE NUMBER	TITLE
None	

ATTACHMENT 8

COLD SHUTDOWN JUSTIFICATION INDEX

CSJ NUMBER	**REV #**	**TITLE**
CSJ-01		Reactor Water Recirculation
CSJ-02		Reactor Building Closed Loop Circulation
CSJ-03		High Pressure Coolant Injection
CSJ-04		*Deleted*
CSJ-05		Main Steam
CSJ-06		Main Steam
CSJ-07		Reactor Water Cleanup
CSJ-08		Main Steam
CSJ-09		Reactor Core Isolation
CSJ-10		Residual Heat Removal
CSJ-11		Core Spray

ATTACHMENT 9

COLD SHUTDOWN JUSTIFICATIONS

Cold Shutdown Justifications

CSJ-01

SYSTEM: **REACTOR WATER RECIRCULATION (RWR)**

COMPONENTS: 02-2MOV-53A, B

CATEGORY: B

SAFETY FUNCTION: These valves close, on low reactor pressure to isolate the faulted loop coincident with initiation of the RHR System in the LPCI mode, to prevent diversion of LPCI flow.

JUSTIFICATION: During normal plant operation, exercising these valves will trip the respective recirculation pump when the valve is 10% open. Securing either pump (single loop operation) is limited by Technical Specification requirements and also requires a reduction in power. This hardship is not warranted since there is no compensating increase in the level of quality and safety.

ALTERNATE TEST: These valves will be stroke time tested during cold shutdown when Reactor Water Recirculation Pumps can be secured in accordance with ISTC-3521(f) and (g).

CSJ-02

SYSTEM: **REACTOR BUILDING CLOSED LOOP COOLING (RBCLC)**

COMPONENTS: 15AOV-130A, B; 15AOV-131A, B; 15AOV-132A, B; 15AOV-133A, B; 15AOV-134A

CATEGORY: A

SAFETY FUNCTION: These valves close to provide containment isolation.

JUSTIFICATION: During normal plant operation, these valves must remain open to provide cooling water to the Drywell coolers, Drywell equipment drain sump cooler, cooling water to the recirculation pump motor and seal coolers. Closing these valves during plant operation could cause a spike in drywell pressure due to the loss of cooling water flow, which may result in a reactor scram and plant shutdown, or damage to the recirculation pumps.

ALTERNATE TEST: These valves will be stroke time tested during cold shutdowns in accordance with ISTC-3521(f) and (g). 15AOV-132A/B and 15AOV-133A/B will be stroke time tested during cold shutdowns when Reactor Water Recirculation Pumps can be secured.

CSJ-03

SYSTEM: **HIGH PRESSURE COOLANT INJECTION (HPCI)**

COMPONENTS: 23HPI-18

CATEGORY: C

SAFETY FUNCTION: This valve opens to provide a flowpath for the HPCI system injection to the reactor vessel.

JUSTIFICATION: With the reactor at operating pressure, the HPCI pump can develop sufficient discharge pressure to open this valve; however HPCI injection of cold water to the reactor vessel during critical operation could result in an undesirable reactivity excursion and thermal transient to the piping components. During plant operation, the differential pressure developed across the valve disc could be in excess of 1000 psid because feedwater pump discharge pressure is present - precluding manual manipulation of the valve. Therefore, this valve cannot be exercised during normal plant operation.

ALTERNATE TEST: This valve will be mechanical exercise tested during cold shutdown in accordance with ISTC-3522(d) and (e).

CSJ-04

DELETED

CSJ-05

SYSTEM: **MAIN STEAM (MSS)**

COMPONENTS: 29AOV-86A, B, C, D

CATEGORY: A

SAFETY FUNCTION: The Main Steam Isolation Valves (MSIV's) are normally open valves that close to isolate containment from the main steam system.

JUSTIFICATION: Exercising these valves during normal operation isolates one line of steam flow to the turbine. Isolation of a main steam header would cause a severe pressure transient in the associated main steam line possibly resulting in a plant trip. Additionally, closure of an MSIV, at power, could potentially result in challenging the setpoint of the main safety relief valves causing inadvertent lifting. Industry experience also indicates that closing the MSIVs under high steam flow conditions may be a contributing factor in observed seat degradation. Seat degradation occurring during valve exercising could result in a loss of primary containment integrity. Therefore, it is impractical to full-stroke exercise these valves to the closed position on a quarterly (nominal 92 days) frequency during plant operation.

The MSIVs have the capability and are being partial stroked during the Technical Specification MSIV scram sensor channel functional test requirements. To completely partially fail-safe exercise these valves to the closed position, the airlines to the valves must be isolated. Thus, with the loss of air, the fail-safe mechanism (springs) would be demonstrated. The resultant exercising of the Main Steam Isolation Valves (MSIV's) could place the plant in an unsafe mode of operation causing transient conditions which could result in a reactor scram. Therefore, partial stroke exercise testing increases the risk of a valve closure when the unit is generating power. This concern was realized within the fleet and the industry and has resulted in full closure of the applicable MSIV and a reactor trip on high pressure.

NUREG-1482 "Guidelines for Inservice Testing at Nuclear Power Plants", Section 2.4.5, "Deferring Valve Testing to Cold Shutdown or Refueling Outages" identifies "impractical conditions justifying test deferrals" as those conditions that could result in unnecessary challenges to safety systems, place undue stress on components, cause unnecessary cycling of equipment, or unnecessarily reduce the life expectancy of the plant systems and components. As such, it is impractical to partially exercise MSIVs on a quarterly (nominal 92 days) frequency during plant operation.

ALTERNATE TEST: These valves will be full-stroke exercise tested to the closed position and fail-safe tested during cold shutdowns per ISTC-3521(c) and (f).

Additional Information to Support Alternative Test

On November 1, 2017, Exelon Generation Company, LLC (Exelon) submitted a relief request associated with the Inservice Testing (IST) programs for Clinton Power Station, Unit 1; Dresden Power Station, Units 2 and 3; James A. FitzPatrick Nuclear Power Plant; LaSalle County Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Oyster Creek Nuclear Generating Station; and Quad Cities Nuclear Power Station, Units 1 and 2.

The relief request submitted on November 1, 2017 proposed an authorization to continue to partial stroke exercise MSIVs on a limited basis with a Cold Shutdown Justification currently in place for MSIVs. Exelon proposed that the partial stroke exercise of MSIVs would be performed in accordance with the Surveillance Frequency Control Program (SFCP) and would partially stroke exercise MSIVs at variant test intervals until the final refueling outage testing interval was achieved. Exelon's relief request was submitted due to the belief that ISTC-3521(b) and ISTC-3521(c) prohibited any type of exercising of MSIVs with a cold shutdown in place.

On February 26, 2018, the NRC held a public meeting to discuss the relief request. The NRC staff stated that the alternative is not needed, since the ASME OM Code does not prevent the continued exercising of MSIVs at power if a CSJ documented in the IST Program Plan document for each site demonstrates that exercising at power is not practicable. In particular, the NRC staff noted that paragraph ISTC-3521 (c) of the ASME OM Code states that exercising "may"- not "shall" - be limited to full-stroke exercising during cold shutdown. Thus, paragraph ISTC-3521 (c) does not prevent the partial-stroke testing of the MSIVs at power. Exelon will utilize paragraph ISTC-3521 (c) for full stroke exercising of the MSIVs during cold shutdown under the IST program while continuing to partially stroke exercise the MSIVs during power operation under the SFCP at various intervals commensurate with the SFCP frequencies.

The following documents are attached to support this CSJ

- NRC Summary of 022618 mtg ML18058A523
- 2018_0327 Fleet IST RR for MSIV Stroke Frequency Withdrawal letter
- NRC withdrawal documentation ML18100A015

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 16, 2018

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION, UNIT NO. 1 ; DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3; JAMES A. FITZPATRICK NUCLEAR POWER PLANT; LASALLE COUNTY STATION, UNITS 1 AND 2; NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2; OYSTER CREEK NUCLEAR GENERATING STATION; AND QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2—WITHDRAWAL OF PROPOSED ALTERNATIVE TO THE MAIN STEAM ISOLATION VALVE TESTING REQUIREMENTS (EPID L-2017-LLR-OI 34)

Dear Mr. Hanson:

By application dated November 1 , 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML 17306A014), Exelon Generation Company, LLC (the licensee) submitted a request in accordance with Paragraph 50.55a(z)(1) of Title 10 of the Code of Federal Regulations (10 CFR) for a proposed alternative to the requirements of 10 CFR 50.55a, "Codes and standards," and the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (0M Code) at Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; James A. FitzPatrick Nuclear Power Plant; LaSalle County Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Oyster Creek Nuclear Generating Station; and Quad Cities Nuclear Power Station, Units 1 and 2. The proposed alternative was intended to allow the licensee to eliminate the quarterly partial-stroke exercise testing of the main steam isolation valves (MSIVs) for each of these facilities. By letter dated March 27, 2018

(ADAMS Accession No. ML18086B221), the licensee requested to withdraw its November 1, 2017, application because the licensee has concluded that the proposed alternative is not needed.

Currently, the licensee performs partial-stroke testing of the MSlVs quarterly and full-stroke testing of the MSlVs during cold shutdown based on a cold shutdown justification (CSJ). Paragraph ISTC-3521 (b) of the ASME 0M Code states that if full-stroke exercising of a valve during operation at power is not practicable, it may be limited to partial-stroke during operation at power and full-stroke during cold shutdown. The application stated that the licensee will revise its CSJ to eliminate the quarterly partial-stroke testing of MSlVs at power, such that the only exercising of MSlVs will be the full-stroke testing during cold shutdown. Paragraph ISTC-3521 (c) of the ASME 0M Code states that if exercising of a valve during operation at power is not practicable, it "may" be limited to full-stroke exercising during cold shutdown.

The licensee also performs a quarterly reactor protection system (RPS) channel functional test (a surveillance requirement) that uses the partial-stroke of the MSIV to generate the MSIV position switch input into the RPS logic. The application stated that the licensee will use the surveillance frequency control program to extend the RPS channel functional test to refueling outage intervals. In order to accomplish this change, the application stated that the partial stroking of the MSlVs at power would need to continue for a number of years at longer intervals (i.e., less frequent than the current quarterly testing).

The application indicated that the licensee's proposed alternative to the ASME 0M Code was needed because continuing to perform periodic partial-stroke MSIV testing at power will contradict a CSJ that exercising at power is not practicable. However, paragraph ISTC-3521 (c) of the ASME 0M Code states that exercising "may"—not "shall"—be limited to full-stroke exercising during cold shutdown. Thus, paragraph ISTC-3521(c) does not prohibit the partial stroke testing of the MSlVs at power. This fact was discussed with the licensee during a public teleconference held on February 26, 2018 (ADAMS Accession No. ML 18058A523). Based on this, the licensee has concluded that the proposed alternative is not needed and has requested to withdraw its November 1, 2017, application.

This letter acknowledges that the licensee has withdrawn its November 1, 2017, application. If you have any questions, please contact Blake Purnell at 301-415-1380 or via e-mail at Blake.Purnell@nrc.gov.

Sincerely,

Blake Purnell, Project Manager
Plant Licensing Branch Ill
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-461 , 50-237, 50-249, 50-333
50-373, 50-374, 50-220, 50-410,
50-219, 50-254, and 50-265

cc: ListServ

B. Hanson - 3 -

DISTRIBUTION.

PUBLIC
RidsRgn1 MailCenter Resource
RidsRgn3MailCenter Resource
RidsNrrDorlLp11 Resource
RidsNrrDorlLp13 Resource
RidsNrrDorlLspb Resource
RidsNrrLASRohrer Resource
RidsNrrLAIBetts Resource
RidsACRS MailCTR Resource
RidsNrrPMExelon Resource
RidsNrrPMClinton Resource
RidsNrrPMDresden Resource
RidsNrrPMFitzPatrick Resource
RidsNrrPMLaSalle Resource
RidsNrrPMNineMilePoint Resource
RidsNrrPMOysterCreek Resource
RidsNrrPMPeachBottom Resource
RidsNrrPMQuadCities Resource
RidsNrrLASRohrer Resource
RidsNrrDeEmib Resource
JBowen, EDO
ITseng, NRR
RW01fgang, NRR
BVaiyda, NRR

ADAMS Accession No. ML18100A015 *via email

OFFICE	LPL3/PM	LPL3/LA	DE/EMIB/BC*	LPL3/BC	LPL3/PM
NAME	BPurnell	SRohrer	SBailey	DWrona	BPurnell
DATE	4/1 1/18	4/11/18	4/9/18	4/16/18	4/16/18

OFFICIAL RECORD COPY

CSJ-06

SYSTEM: **MAIN STEAM (MSS)**

COMPONENTS: 29MOV-203A, B

CATEGORY: B

SAFETY FUNCTION: These valves open to provide flow paths for post-accident MSIV packing leak-off to the Standby Gas Treatment System.

JUSTIFICATION: Opening these valves during power operation could subject downstream piping to pressures in excess of its 150 psig design pressure.

ALTERNATE TEST: These valves will be stroke time tested during cold shutdown in accordance with ISTC-3521(f) and (g).

CSJ-07

SYSTEM: **REACTOR WATER CLEANUP**

COMPONENTS: 12MOV-15, 12MOV-18, 12MOV-69

CATEGORY: A

SAFETY FUNCTION: These valves close to provide containment isolation. The valves also close on, low reactor water level or high RWCU ambient temperature to protect the core in case of a break in the RWCU piping and on SLC actuation to prevent removal of boron.

JUSTIFICATION: Cycling these valves during operation has significant negative effects to reactor water chemistry that could result in power reduction or plant shutdown. Radiation exposure received during system alterations to perform the testing during operation has also resulted in excessive personnel exposure. Cycling the system during operation causes thermal transients that places undue stress on the piping and pumps. Testing of these valves during operation subjects the system to unacceptable chemical and thermal transients and excessive personnel radiation exposure. As discussed in NUREG-1482 Paragraph 2.4.5 these negative effects place impractical conditions on the system and justify cold shutdown deferral.

ALTERNATE TEST: These valves will be stroke time tested during cold shutdown in accordance with ISTC-3521(f) and (g).

CSJ-08

SYSTEM:	**MAIN STEAM (MSS)**
COMPONENTS:	29AOV-80A, B, C, D; 29AOV-86A, B, C, D
CATEGORY:	A
SAFETY FUNCTION:	These valves close to provide containment isolation.
JUSTIFICATION:	Full stroke testing of MSIV's at power places the plant in an abnormal operating condition and introduces an unnecessary challenge to plant equipment. This is in view of industry experience, both from an operational standpoint, and from the standpoint that stroking MSIV's at power is a contributor to valve seat degradation and resultant degraded containment isolation capability. (Ref: NUREG-1482)
ALTERNATE TEST:	Stroke timing during cold shutdown in accordance with ISTC-3521(f) and (g) is acceptable since valve actuator is designed to limit stroke time regardless of system dynamics present at time of testing.

CSJ-09

SYSTEM: **REACTOR CORE ISOLATION COOLING (RCIC)**

COMPONENTS: 13RCIC-22

CATEGORY: C

SAFETY FUNCTION: This valve opens to provide a flow path for the RCIC system injection to the reactor vessel.

JUSTIFICATION: With the reactor at operating pressure, the RCIC pump can develop sufficient discharge pressure to open this valve, however RCIC injection of cold water to the reactor vessel during critical operation could result in an undesirable reactivity excursion and thermal transient to the piping components. During plant operation, the differential pressure developed across the valve disc could be in excess of 1000 psid - precluding manual manipulation of the valve. Therefore, this valve cannot be exercised during normal plant operation.

ALTERNATE TEST: This valve will be mechanical exercise tested during cold shutdown in accordance with ISTC-3522(d) and (e).

CSJ-10

SYSTEM:	**RESIDUAL HEAT REMOVAL (RHR)**
COMPONENTS:	10MOV-25A and B
CATEGORY:	A
SAFETY FUNCTION:	These valves open to provide a flow path for the RHR system injection to the reactor vessel. These valves provide the pressure isolation function to protect the low-pressure interconnecting RHR piping from the high pressure Reactor Coolant system.
JUSTIFICATION:	These pressure isolation motor-operated valves maintain one of the two high to low pressure barriers during plant operation. The other pressure isolation barrier is a check valve. Opening a PIV with only one other PIV available, especially with it being a check valve, increases the probability of over-pressurizing the low-pressure core spray system and inter-system LOCA. These valves are not designed to open with high differential pressure across the seats. To exercise these valves during plant operation requires installation of hydrostatic test equipment and instrumentation to equalize pressure across the valve seats when the reactor is pressurized.
ALTERNATE TEST:	Exercise test during cold shutdown.

CSJ-11

SYSTEM:	**CORE SPRAY (CS)**
COMPONENTS:	14MOV-12A and B
CATEGORY:	A
SAFETY FUNCTION:	These valves open to provide a flow path for the CS system injection to the reactor vessel. These valves provide the pressure isolation function to protect the low-pressure interconnecting CS piping from the high pressure Reactor Coolant system.
JUSTIFICATION:	These pressure isolation motor-operated valves maintain one of the two high to low pressure barriers during plant operation. The other pressure isolation barrier is a check valve. Opening a PIV with only one other PIV available, especially with it being a check valve, increases the probability of over-pressurizing the low-pressure core spray system and inter-system LOCA. The valves are not designed to open with high differential pressure across the seats and instrumentation is not installed to determine the differential pressure prior to stroking the valve from open to close. Installation of test equipment would be required to conduct the testing quarterly.
ALTERNATE TEST:	Exercise test during cold shutdown.

ATTACHMENT 10

REFUELING OUTAGE JUSTIFICATION INDEX

ROJ NUMBER	TITLE
ROJ-01	*Deleted*
ROJ-02	Reactor Water Recirculation
ROJ-03	Reactor Water Recirculation
ROJ-04	Automatic Depressurization
ROJ-05	Residual Heat Removal
ROJ-06	Residual Heat Removal
ROJ-07	Standby Liquid Control
ROJ-08	Reactor Core Isolation Cooling
ROJ-09	Core Spray
ROJ-10	Core Spray
ROJ-11	*Deleted*
ROJ-12	High Pressure Coolant Injection
ROJ-13	High Pressure Coolant Injection
ROJ-14	High Pressure Coolant Injection
ROJ-15	High Pressure Coolant Injection
ROJ-16	High Pressure Coolant Injection
ROJ-17	High Pressure Coolant Injection
ROJ-18	High Pressure Coolant Injection
ROJ-19	Main Steam
ROJ-20	Feedwater
ROJ-21	Instrument Air
ROJ-22	*Deleted*
ROJ-23	High Pressure Coolant Injection
ROJ-24	Reactor Core Isolation Cooling
ROJ-25	Reactor Core Isolation Cooling
ROJ-26	High Pressure Coolant Injection
ROJ-27	Control Rod Drive Hydraulics
ROJ-28	Residual Heat Removal
ROJ-29	Residual Heat Removal
ROJ-30	Feedwater
ROJ-31	RCIC and HPIC
ROJ-32	Containment Atmosphere Dilution
ROJ-33	Reactor Water Recirculation
ROJ-34	Radioactive Waste
ROJ-35	Main Steam

ATTACHMENT 11

REFUELING OUTAGE JUSTIFICATIONS

ROJ-01

DELETED. JAF has an approved relief request. A refueling outage justification is not necessary for excess flow check valve testing.

ROJ-02

SYSTEM:	**REACTOR WATER RECIRCULATION (RWR)**
COMPONENTS:	02-2RWR-13A, B
CATEGORY:	A/C
SAFETY FUNCTION:	These recirculation pump seal water injection valves close to provide containment isolation.
JUSTIFICATION:	Exercising these valves during normal operations or cold shutdown requires securing the Recirculation pumps and entering containment to check the valves closed by using a back-leakage test. Testing during operations is therefore impossible. Testing during cold shutdown by performing back-leakage tests would require extensive time for test equipment set-up and place an undue burden on the plant staff. In addition, entry into the containment may be prohibited if the drywell remains inerted.
ALTERNATE TEST:	Back-leakage testing and leak rate testing will be performed during each refueling outage in accordance with ISTC-3522(c) and (f).

ROJ-03

SYSTEM:	**REACTOR WATER RECIRCULATION (RWR)**
COMPONENTS:	02-2RWR-41A, B
CATEGORY:	A/C
SAFETY FUNCTION:	These recirculation pump seal purge check valves close to provide containment isolation.
JUSTIFICATION:	Closing these valves any time Reactor Water Recirculation Pumps are running subjects the pump seals to thermal transients and pressure fluctuations, thereby, shortening seal life. Pressure fluctuations and oscillations can degrade the pressure-retaining ability of either or both seal stages. Additionally, securing seal purge flow while the Reactor Water Recirculation Pumps are running introduces reactor coolant and associated corrosion products into the seal cavity, which also shortens seal life.
ALTERNATE TEST:	Back-leakage testing and leak rate testing will be performed during each refueling outage in accordance with ISTC-3522(c) and (f).

ROJ-04

SYSTEM: **AUTOMATIC DEPRESSURIZATION (ADS)**

COMPONENTS: 02RV-1 through 02RV-11
02VB-1 through 02VB-11

CATEGORY: C

SAFETY FUNCTION: These valves remain closed to prevent steam from an open safety/relief valve (SRV) from entering the drywell. They open following closure of an SRV to prevent the formation of a water column within the downcomer that could cause torus damage during subsequent lifting of the same SRV.

JUSTIFICATION: Exercising these valves requires local manipulation of each valve and thus entry into the containment. During plant operation at power, and on occasion while in cold shutdown, the containment atmosphere is maintained in a nitrogen-inerted condition. During such periods, entry into the containment is not practical due to personnel safety concerns.

ALTERNATE TEST: Testing will be performed during each refueling outage in accordance with ISTC-3522(c) and (f).

ROJ-05

SYSTEM: **RESIDUAL HEAT REMOVAL (RHR)**

COMPONENTS: 10RHR-64A, B, C, D

CATEGORY: C

SAFETY FUNCTION: These valves open on forward flow to provide minimum flow protection for the RHR pumps and close on reverse flow to prevent diversion of flow through an idle parallel pump.

JUSTIFICATION: These valves are exercised open every three months by flow during pump testing. However, quantitative flow measurements as a means of verifying these valves open has been determined to be impractical.

There is no installed flow instrumentation in the minimum flow line thus attempts at flow measurements are being made with a strap on ultrasonic flow meters. Due to the minimum flow line configuration and operating conditions, there is a high amount of cavitation/turbulence in the line causing the ultrasonic flow meter to go into fault. Attempts have been made at different locations and with different size transducers and faults still occur.

This test method requires the RHR pumps to be operated repeatedly (three to four times) at minimum flow conditions for the maximum time period allowed by procedure. Running at this condition is undesirable, particularly for a test method that frequently does not yield meaningful results. NRC Information Notice 89-08 documented concerns about pump damage by operating at low flow conditions. When this test is performed with no flow measurements being taken, the time spent at minimum pump flow is short.

In addition, this testing must be performed in a radiation area, which has caused increased exposure to personnel while multiple test attempts and transducer repositioning are accomplished. It is concluded that continued efforts with this method are not practical.

Attempts were made to distinguish the check valve opening impact on the valve bonnet using a seismic vibration probe. Meaningful results could not be obtained again due to the high background noise and vibration associated with a pump start at minimum flow.

The method of using process flow and pressure instrumentation in the main line to infer the flow in the minimum flow line was investigated. However, the small flow rate through the minimum flow line in comparison with the main line flow would not be discernable within the accuracy of the process instrumentation.

ALTERNATE TEST: In accordance with ISTC-5221(c), a sample disassembly examination program will be implemented for this Group of valves. During each refuel outage at least one (1) valve will be disassembled, inspected, and verified operable. The acceptance criteria as stated in ISTC-5221(c)(2) is provided in the maintenance procedure used for check valve disassembly. If any valve is found to be inoperable, the remaining valves will be disassembled and inspected prior to startup. The inspection schedule will be such that all four (4) valves in the group are inspected at least once every eight (8) years. These valves will be full or part-stroke exercised, if practicable, after reassembly.

ROJ-06

SYSTEM: **RESIDUAL HEAT REMOVAL (RHR)**

COMPONENTS: 10RHR-95A,B

CATEGORY: C

SAFETY FUNCTION: These valves close to prevent reverse flow from the torus.

JUSTIFICATION: These are simple check valves with no means of determining disc position without performing a back leakage test. Performing such a test during plant operations would require setting up a test rig and performing a hydrostatic test. As discussed in NUREG 1482, Revision 2, the NRC has determined that the need to set up test equipment is adequate justification to defer backflow testing of a check valve until a refueling outage.

During cold shutdown, the system lineup changes and the effort involved with setting up test equipment would constitute an unreasonable burden on the plant staff.

ALTERNATE TEST: These valves will be verified to close each refueling outage during a hydrostatic leak rate test in accordance with ISTC-3522(c) and (f).

ROJ-07

SYSTEM: **STANDBY LIQUID CONTROL (SLC)**

COMPONENTS: 11SLC-16 & 11SLC-17

CATEGORY: A/C

SAFETY FUNCTION: These valves prohibit backflow from the reactor vessel to the SLC System and provide for containment isolation. They open to permit SLC System flow to the reactor vessel.

JUSTIFICATION: Full-stroke exercising these valves requires that flow be established through the subject check valves. The only practical means of initiating flow through these valves requires actuation of the SLC system and pumping from the SLC Tank to the reactor vessel. During normal plant operation, this would introduce boron into the reactor vessel resulting in unacceptable reactivity and chemistry transients. Testing during cold shutdown would result in chemistry transients and undue burden on the plant staff with respect to maintenance of the SLC pump explosive valves.

ALTERNATE TEST: Testing will be conducted during each refueling outage and as required by Technical Specifications, by injecting water into the reactor vessel by use of the Standby Liquid Control pumps.

ROJ-08

SYSTEM: **REACTOR CORE ISOLATION COOLING (RCIC)**

COMPONENTS: 13RCIC-04 and 13RCIC-05

CATEGORY: A/C

SAFETY FUNCTION: These valves close to provide containment isolation.

JUSTIFICATION: There is no provision on either of these valves that provides position indication of the disc. As a result, valve closure must be verified by back-leakage testing. In order to verify valve closure by the back-leakage technique, the RCIC exhaust line must be isolated for the duration of the test causing the RCIC system to be inoperable.

The potential safety impact of voluntarily placing the RCIC system in an inoperable status during plant operation at power is considered to be imprudent and unwarranted in relation to any apparent gain in system reliability derived from the closure verification. In addition, the valves are located approximately twenty (20) feet from the floor necessitating erection of a large scaffold in the vicinity of the RCIC pump. This also is considered to be undesirable from the aspect of potential damage to RCIC system components should the scaffold be subjected to structural failure.

Based on the foregoing discussion, testing of these valves during plant operation at power is considered to be impractical. During cold shutdowns, erection of the scaffold in addition to other activities related to test performance would place an extreme burden on the plant staff and would likely result in unwarranted extensions to all forced outages with the added negative impact on plant performance and availability.

ALTERNATE TEST: These valves will be verified to close by performing a back-leakage test at each refueling outage in accordance with ISTC-3522(c) and (f).

ROJ-09

SYSTEM: **CORE SPRAY (CSP)**

COMPONENTS: 14AOV-13A,B

CATEGORY: A/C

SAFETY FUNCTION: These valves open to provide flow paths from the Core Spray System to the reactor vessel. They close for pressure isolation protection of the low pressure core spray piping.

JUSTIFICATION: There is no mechanism by which these valves can be full-stroke exercised without injecting water from the core spray pumps to the reactor vessel. During plant operation, the core spray pumps cannot produce sufficient discharge pressure to overcome reactor vessel pressure and provide flow into the vessel.

The installed air operators are capable of exercising the valves, providing there is not differential pressure across the valve seat. During plant operation, there is a significant differential pressure across the valve seat.

During cold shutdown, injecting into the reactor vessel requires a major effort to establish the prerequisite conditions and realignment of the Core Spray system to allow supplying water from the Condensate Storage Tank. Torus water cannot be used since it does not meet the chemistry requirements for reactor grade makeup. It is estimated that such a test would take about 24 hours to perform and would result in a significant burden on the plant operating staff. In addition, there is a potential for overfilling the reactor vessel and flooding the main steam lines. This could adversely affect the performance of the main steam safety/relief valves (SRVs) since a contributing factor to the historically poor performance of the SRVs is water contamination of the operators.

ALTERNATE TEST: During refueling outages, the valves will be exercised using a mechanical exerciser. In addition, the valves require pressure isolation valve leak rate tests once every two years.

Each of the valves will be full-stroked exercised during each refuel outage per ISTC-5221(b) and ISTC-3521(c) and (f) and leak rate tested once every two years per ISTC-3630(a) to satisfy full open and close exercising tests.

ROJ-10

SYSTEM: **CORE SPRAY (CSP)**

COMPONENTS: 14CSP-62A,B

CATEGORY: C

SAFETY FUNCTION: These valves close to prevent reverse flow from the torus.

JUSTIFICATION: There are no position indicators or other means to verify closure of these valves. As a result, valve closure must be verified by back-leakage testing. Performing such a test during plant operations would require setting up for and performing a hydrostatic test. As discussed in NUREG 1482, Revision 2, section 4.1.6, the NRC has determined that the need to set up test equipment is adequate justification to defer backflow testing of a check valve until a refueling outage. During cold shutdown, the system lineup changes and the effort involved with setting up test equipment would constitute an unreasonable burden on the plant staff.

ALTERNATE TEST: These valves will be verified close each refueling outage in accordance with ISTC-3522(c) and (f) during a hydrostatic leak rate test.

ROJ-11

DELETED

ROJ-12

SYSTEM: **HIGH PRESSURE COOLANT INJECTION (HPCI)**

COMPONENTS: 23HPI-12 and 23HPI-65

CATEGORY: A/C

SAFETY FUNCTION: These valves close to provide containment isolation.

JUSTIFICATION: There is no provision on either of these valves that provides position indication of the disc. As a result, valve closure must be verified by back-leakage testing. In order to verify valve closure by the back-leakage technique, the HPCI exhaust line must be isolated for the duration of the test causing the HPCI system to be inoperable. The potential safety impact of voluntarily placing the HPCI system in an inoperable status during plant operation at power is considered to be imprudent and unwarranted in relation to any apparent gain in system reliability derived from the closure verification. In addition, the valves are located approximately twenty (20) feet from the floor necessitating erection of a large scaffold in the vicinity of the HPCI pump. This also is considered to be undesirable from the aspect of potential damage to HPCI system components should the scaffold be subjected to structural failure.

Based on the foregoing discussion, testing of these valves during plant operation at power is considered to be impractical. During cold shutdowns, erection of the scaffold in addition to other activities related to test performance would place an extreme burden on the plant staff and would likely result in unwarranted extensions to all forced outages with the added negative impact on plant performance and availability.

ALTERNATE TEST: These valves will be verified to close by performing a back-leakage test at each refueling outage in accordance with ISTC-3522(c) and (f).

<u>ROJ-13</u>

SYSTEM: **HIGH PRESSURE COOLANT INJECTION (HPCI)**

COMPONENTS: 23HPI-13 and 23HPI-56

CATEGORY: C

SAFETY FUNCTION: These valves open to permit HPCI turbine condensate to drain to the Torus and close on cessation of flow.

JUSTIFICATION: There are no means for exercising these valves to the open position where positive indication of acceptable valve performance is verified. There is no provision that provides position indication of the disc. There are no test taps and block valves to enable a back-leakage test to verify closure.

ALTERNATE TEST: ISTC-5221(c) allows a sample disassembly program such that one valve is disassembled and inspected each refueling outage to verify operability as an alternative to quarterly testing. The grouping requirements of ISTC-5221(c) shall be followed. These valves will be full or part-stroke exercised, if practicable, after reassembly.

ROJ-14

SYSTEM: **HIGH PRESSURE COOLANT INJECTION (HPCI)**

COMPONENTS: 23HPI-32

CATEGORY: C

SAFETY FUNCTION: This valve closes during the suction swap from the Condensate Storage Tank to the torus to prevent diversion of the torus flow from the HPCI pump suction.

JUSTIFICATION: There is no provision on this valve that provides position indication of the disc. There are no block valves between this valve and the suction of the HPCI pump to enable a back-leakage test to verify closure.

ALTERNATE TEST: ISTC-5221(c) allows sample disassembly program each refueling outage to verify operability as an alternative to quarterly testing. This valve will be full or part-stroke exercised, if practicable, after reassembly.

ROJ-15

SYSTEM:	**HIGH PRESSURE COOLANT INJECTION (HPCI)**
COMPONENTS:	23HPI-61
CATEGORY:	C
SAFETY FUNCTION:	This valve opens to provide a flow path from the torus to the suction of the HPCI booster pump. It closes on cessation of flow.
JUSTIFICATION:	The only practical method available to full flow exercise this valve is to pump water from the torus into the reactor vessel. Due to the lack of suitable water quality in the torus, this option is not practical. There is no provision on this valve that provides position indication of the disc. There are no test taps and block valves to enable a back-leakage test to verify closure.
ALTERNATE TEST:	ISTC-5221(c) allows a sample disassembly program each refueling outage to verify operability as an alternative to quarterly testing. This valve will be full or part-stroke exercised, if practicable, after reassembly.

ROJ-16

SYSTEM: **HIGH PRESSURE COOLANT INJECTION (HPCI)**

COMPONENTS: 23HPI-62

CATEGORY: C

SAFETY FUNCTION: This valve opens to provide a flow path for minimum flow from the HPCI main pump. It closes on cessation of flow.

JUSTIFICATION: Due to the configuration of the minimum flow motor operated valve control logic, fully developed flow cannot be achieved through this check valve. Additionally, full-stroke exercising cannot be verified with existing instrumentation. There is no provision on this valve that provides position indication of the disc. There are no test taps and block valves to enable a back-leakage test to verify closure.

ALTERNATE TEST: ISTC-5221(c) allows a sample disassembly program each refueling outage to verify operability as an alternative to quarterly testing. This valve will be full or part-stroke exercised, if practicable, after reassembly.

ROJ-17

SYSTEM: **HIGH PRESSURE COOLANT INJECTION (HPCI)**

COMPONENTS: 23HPI-130

CATEGORY: C

SAFETY FUNCTION: This valve opens to provide a flow path for cooling water circulation through the HPCI turbine lube oil cooler and closes to prevent flow diversion.

JUSTIFICATION: This valve has no means of determining disc position or flow rate and, thus there is no mechanism for verifying full accident flow. In addition, there are no test taps and block valves to enable a back-leakage test to verify closure.

ALTERNATE TEST: ISTC-5221(c) allows a sample disassembly program each refueling outage to verify operability as an alternative to quarterly testing. This valve will be full or part-stroke exercised, if practicable, after reassembly.

ROJ-18

SYSTEM: **HIGH PRESSURE COOLANT INJECTION (HPCI)**

COMPONENTS: 23HPI-131

CATEGORY: C

SAFETY FUNCTION: This valve closes to prevent flow diversion from the HPCI booster pump.

JUSTIFICATION: There is no provision on this valve that provides position indication of the disc. There are no test taps and block valves to enable a back-leakage test to verify closure.

ALTERNATE TEST: ISTC-5221(c) allows a sample disassembly program each refueling outage to verify operability as an alternative to quarterly testing. This valve will be full or part-stroke exercised, if practicable, after reassembly.

ROJ-19

SYSTEM: **MAIN STEAM (MSS)**

COMPONENTS: 29AOV-80A, B, C, D

CATEGORY: A

SAFETY FUNCTION: These valves are normally open to provide steam to the main turbine generator and auxiliaries, and they close to isolate steam flow and for containment isolation.

JUSTIFICATION: Fail safe exercising these valves requires local manipulation of valves located inside containment. During plant operation at power, and on occasion while in cold shutdown, the containment atmosphere is maintained in a nitrogen-inerted condition. During such periods, entry into the containment is not practical due to personnel safety concerns.

ALTERNATE TEST: These valves will be verified to fail safe close at each refueling outage in accordance with ISTC-3521(e) and (h).

ROJ-20

SYSTEM: **FEEDWATER (FWS)**

COMPONENTS: 34FWS-28A, B

CATEGORY: A/C

SAFETY FUNCTION: These valves close to provide containment isolation upon cessation of feedwater flow during accident conditions.

JUSTIFICATION: There is no provision on either of these valves that provides position indication of the disc. As a result, valve closure must be verified by back-leakage testing. During plant operation at power, these valves cannot be closed without precipitating a plant shutdown.

During cold shutdowns, performing a back-leakage test requires entry into the containment vessel and extensive system preparations, including draining of the main feedwater piping from the outlet of the sixth point feedwater heaters to the reactor vessel isolation valves (approximately 2000 gallons per line). Furthermore, testing of 34FWS-28B requires shutdown of the cleanup system. It is estimated that testing either of these valves would require up to 24 hours and demand significant staff resources. Also, entry into the containment at cold shutdown with the containment inerted is a personnel safety concern.

ALTERNATE TEST: Closure of these valves will be demonstrated during each refuel outage in accordance with ISTC-3522(c) and (f) by conducting a back-leakage test.

ROJ-21

SYSTEM: **INSTRUMENT AIR (IAS)**

COMPONENTS: 39IAS-22 & 39IAS-29

CATEGORY: A/C

SAFETY FUNCTION: These valves open to provide nitrogen to the MSIVs and the SRV accumulators inside the containment. They close for containment isolation.

JUSTIFICATION: Exercising these valves open is performed by charging the bleed-down header following MSIV testing. During plant operation at power, this is impractical since closure of the MSIVs would cause a plant trip. Also performing such a test requires entry into the containment vessel and local manipulation of test connections located inside the drywell.

During plant operation at power and, on occasion, while in the cold shutdown mode, the containment atmosphere is maintained in a nitrogen-inerted condition. During such periods, entry into the containment is not practical due to personnel safety concerns.

ALTERNATE TEST: These valves will be tested open at each refueling outage in accordance with ISTC-3522(c) and (f).

ROJ-22

DELETED

ROJ-23

SYSTEM: **HIGH PRESSURE COOLANT INJECTION (HPCI)**

COMPONENTS: 23HPI-402 and 23HPI-403

CATEGORY: C

SAFETY FUNCTION: These valves open to eliminate any differential pressure that could force water from the suppression chamber into the HPCI exhaust piping when the suppression chamber pressure is greater than atmospheric. They close to prevent HPCI exhaust steam from entering the suppression chamber air space, thus bypassing the quenching action of the torus.

JUSTIFICATION: Operation of the HPCI pump turbine does not prove operability of these valves and special testing is required. This testing necessitates isolation of the vacuum breaker piping, which results in the inoperability of the HPCI system for the duration of the test. Due to the importance of the HPCI system function and the lack of a redundant HPCI train, to perform this testing during plant operation at power, is considered to be impractical without a compensating level of quality and safety.

Performing this test during cold shutdown requires mobilization of test equipment and as such constitutes adequate justification to defer check valve testing to a refueling outage.

ALTERNATE TEST: These valves will be forward and reverse flow tested each refueling outage in accordance with ISTC-3522(c) and (f).

ROJ-24

SYSTEM: **REACTOR CORE ISOLATION COOLING (RCIC)**

COMPONENTS: 13RCIC-37 & 13RCIC-38

CATEGORY: C

SAFETY FUNCTION: These valves open to eliminate any differential pressure that could force water from the suppression chamber into the RCIC steam exhaust piping when the suppression chamber pressure is greater than atmospheric.

JUSTIFICATION: Verifying proper operation of these valves involves a test that requires isolation of the vacuum breakers for an extended period of time. During this test, the RCIC system is considered to be inoperable. Due to operational concerns associated with the plant's response to possible transients without an operable RCIC system, it is considered to be impractical without a compensating level of quality and safety.

Performing this test during cold shutdown requires mobilization of test equipment and as such constitutes adequate justification to defer check valve testing to a refueling outage.

ALTERNATE TEST: These valves will be forward and reverse flow tested each refueling outage in accordance with ISTC-3522(c) and (f).

ROJ-25

SYSTEM: **REACTOR CORE ISOLATION COOLING (RCIC)**

COMPONENTS: 13RCIC-7

CATEGORY: C

SAFETY FUNCTION: This valve opens to allow condensate drainage from the steam exhaust piping to the suppression chamber. It closes for containment isolation.

JUSTIFICATION: Closure verification for this valve is accomplished by performing a back flow test where the drain line is isolated from the steam exhaust line. Placing the RCIC system in this configuration during plant operation is undesirable and could adversely affect the plant's response in the event of a transient. Open exercise includes similar configuration.

Performing this test during cold shutdown requires mobilization of test equipment and as such constitutes adequate justification to defer check valve testing to a refueling outage.

ALTERNATE TEST: This valve will be reverse flow tested during refuel outages in accordance with ISTC-3522(c) and (f).

ROJ-26

SYSTEM: **HIGH PRESSURE COOLANT INJECTION (HPCI)**

COMPONENTS: 23HPI-13

CATEGORY: C

SAFETY FUNCTION: This valve opens to allow condensate drainage from the steam exhaust piping to the suppression chamber. It closes for containment isolation.

JUSTIFICATION: Closure verification for this valve is accomplished by performing a back flow test where the drain line is isolated from the steam exhaust line and the torus is vented to atmosphere. Placing the HPCI system and containment in this configuration during plant operation could adversely affect the plant's response in the event of an accident and is considered to be impractical without a compensating level of quality and safety.

Performing this test during cold shutdown requires mobilization of test equipment and as such constitutes adequate justification to defer check valve testing to a refueling outage.

ALTERNATE TEST: This valve will be reverse flow tested during refuel outages in accordance with ISTC-3522(c) and (f).

ROJ-27

SYSTEM:	**CONTROL ROD DRIVE HYDRAULICS (CRD)**
COMPONENTS:	03HCU-115 (Typical for 137 HCUs)
CATEGORY:	C
SAFETY FUNCTION:	These valves close on initiation of a scram to prevent diversion of scram drive water into a depressurized charging header.
JUSTIFICATION:	Exercising these valves during operation would require depressurization of the charging header with the potential for a loss of scram function. Performing this test during cold shutdown requires mobilization of test equipment and as such constitutes adequate justification to defer check valve testing to a refueling outage.
ALTERNATE TEST:	These valves will be reverse flow tested during refuel outages in accordance with ISTC-3522(c) and (f).

ROJ-28

SYSTEM: **RESIDUAL HEAT REMOVAL (RHR)**

COMPONENTS: 10MOV-17 & 10MOV-18

CATEGORY: A

SAFETY FUNCTION: These valves remain closed to protect the RHR System piping and components from over-pressurization during plant operation and inadvertent drain down events while in cold shutdown. 10MOV-17 also performs a containment isolation function.

JUSTIFICATION: With the reactor pressure greater than 75 psig, these valves are prevented from opening by an electrical interlock.

ALTERNATE TEST: These valves will be stroke time tested during refuel outages in accordance with ISTC-3521(e) and (h).

ROJ-29

SYSTEM: **RESIDUAL HEAT REMOVAL (RHR)**

COMPONENTS: 10AOV-68A, B

CATEGORY: A/C

SAFETY FUNCTION: These valves open to provide flow paths for LPCI injection to the reactor vessel. They close for pressure isolation from the reactor vessel.

JUSTIFICATION: With the reactor at operating pressure, the RHR pumps cannot develop sufficient discharge pressure to open these valves. The installed air operators are designed to open these valves at zero differential pressure, which is not practical with the reactor at operating pressure. Therefore, these valves cannot be full or part stroke exercised during normal plant operation.

Since there is no position indication for these valves, closure verification must be done by backflow testing. Such testing during plant operation is impractical due to personnel safety concerns related to the potential release of radioactive steam at high pressure.

ALTERNATE TEST: In accordance with recommendations of NUREG-1482, Revision 2, Section 4.1.6, these valves will be forward and reverse flow tested during refueling outages in accordance with ISTC-3522(c) and (f).

ROJ-30

SYSTEM: **FEEDWATER (FWS)**

COMPONENTS: 34NRV-111A, B

CATEGORY: A/C

SAFETY FUNCTION: These valves close to provide containment isolation and to prevent diversion of HPCI flow into the feedwater system.

JUSTIFICATION: Exercising these valves during operation would require isolation of feedwater flow to the reactor vessel. Such an evolution would create an adverse operating condition and potential automatic plant shutdown. To perform this testing during plant operation is considered to be impractical without a compensating level of quality and safety.

ALTERNATE TEST: In accordance with recommendations of NUREG-1482, Revision 2, Section 4.1.6, these valves will be forward and reverse flow tested during refueling outages in accordance with ISTC-3522(c) and (f).

ROJ-31

SYSTEM: **Reactor Core Isolation Cooling and High Pressure Coolant Injection**

COMPONENT: 13MOV-15, 23MOV-15

CATEGORY: A

SAFETY FUNCTIONS: 13MOV-15 must close to provide containment isolation and also closes in the event of a RCIC steam line break. RCIC is not a credited ECCS/ESF system; therefore, the 13MOV-15 open function is not a safety function.

23MOV-15 must close to provide containment isolation and also closes in the event of a HPCI steam line break. This valve has an open safety function to supply steam to the HPCI turbine.

TEST REQUIREMENTS: Full stroke exercise and stroke time test quarterly.

JUSTIFICATION: The valves are located in the inerted containment during power operation and are inaccessible. These valves would cause a loss of system function requiring RCIC or HPCI to be inoperable if they were to fail closed during a quarterly test. A unit shutdown would be required to perform corrective maintenance.

The ASME OM Code-of-Record allows licensees to perform testing during cold shutdown if it is not practical to test such valves during power operation. Similarly, the ASME OM Code allows licenses to test valves during each refueling outage if it is impractical to test the valve during cold shutdowns. The staff has determined that it is impractical to de-inert the containment during each cold shutdown outage solely to perform such routine testing or repair activities.

ALTERNATE TEST: Full stroke exercise and stroke time test each refueling.

REFERENCES: NUREG-1482, Revision 2, Paragraph 3.1.1 Deferring Valve Testing to Each Cold Shutdown or Refueling Outage

NUREG-1482, Revision 2, 3.1.1.3 De-inerting Containment of Boiling-Water Reactors To Allow Cold Shutdown Testing

ROJ-32

SYSTEM:	**Containment Atmosphere Dilution**
COMPONENT:	27CAD-67, 68, 69 and 70
CATEGORY:	A/C
SAFETY FUNCTIONS:	These valves have a safety function in the open position to provide nitrogen dilution to primary containment to maintain hydrogen and oxygen concentrations below explosive limits. These valves have a safety function in the closed position to maintain primary containment integrity.
TEST REQUIREMENTS:	Reverse closure test quarterly
ALTERNATE TEST:	Closure test by Appendix J LLRT each refueling outage.
JUSTIFICATION:	There are no position indicators or other means to verify closure of these valves. During normal plant operation, test equipment to conduct a pressure drop test or local leak rate test must be installed to verify disk position. Valve closure will be verified by local leak rate testing each refueling outage. When it is impracticable to verify check valve closure during plant operation or cold shutdown, it is acceptable to extend the check valve quarterly exercise test (both open and close) to the refueling outage when the closure verification may be performed in conjunction with the Type C leak rate test. The open exercise test may also be performed during the refueling outage or anytime during the fuel cycle interval.
REFERENCES:	NUREG-1482 Rev. 2, section 4.1.6.

ROJ-33

SYSTEM: **Reactor Water Recirculation (RWR)**

COMPONENT: 02-2AOV-39 Recirc Loop Inboard Sample Isolation Valve

CATEGORY: A

TEST REQUIREMENTS: Quarterly Exercise /Stroke Time Close and Fail-Safe testing

SAFETY FUNCTION: The recirculation loop sample isolation valve closes to provide containment isolation. The valve has an open function to allow sampling of the water by chemistry.

Deferred Testing: ISTC-3510, 3560, & 5121: Quarterly Exercise /Stroke Time Close and Fail Safe testing

JUSTIFICATION: The valve is located in the inerted containment and is inaccessible. In addition, entry into the containment during cold shutdown may be prohibited if the drywell remains inerted.

This valve would cause a loss of the open function (chemistry sampling) if it was to fail closed during a quarterly test. A unit shutdown would be required to perform corrective maintenance

Although this valve is capable of being tested at power, operating experience at other Exelon and non-Exelon plants has demonstrated a potential for valves to fail or degrade because of cycling at power (Reference ICES #314926, #314808 and #305581). Failure modes have included severe packing leakage and a loss of containment isolation function. Due to the fact that this valve is located inside the drywell and is "inaccessible," the inability of the valve to open or close would result in a degraded system with the potential for an unnecessary plant shutdown and challenge to safety systems. Additionally, the containment would require de-inerting in order to perform repairs.

Thus the risks associated with testing this valve on line vs during refuel outages* outweigh the benefits of testing quarterly on line. During refuel outages, the primary containment is open and any repairs could be performed immediately without cycling the plant and challenging safety systems during the process of shutting down the plant. Thus this analysis provides the basis, in accordance with ISTC-3521(e) for deferring testing from a quarterly test frequency to refuel outages as noted in NUREG 1482 section 2.4.5 "Deferring Valve Testing to Cold Shutdown or Refueling Outages".

* It has been determined under this evaluation that testing during cold shutdowns (ref ISTC-3521(c) is not practicable because primary containment (drywell and suppression chambers) is not always made accessible.

REFERENCES: NUREG-1482 Revision 2, Paragraph 3.1.1 Deferring Valve Testing to Each Cold Shutdown or Refueling Outage

NUREG-1482 Revision 2, Paragraph 3.1.1.3 De-inerting Containment if Boiling-Water Reactors to Allow Cold Shutdown Testing

ROJ-34

SYSTEM:	**Radioactive Waste**
COMPONENTS:	20MOV-82 Drywell Floor Drain Sump Pump Discharge Inboard Isolation Valve 20MOV-94 RDW Drywell Equipment Drain Sump Pump Discharge Inboard Isolation Valve
CATEGORY:	A
TEST REQUIREMENTS:	Quarterly Exercise & Stroke Time Close testing
SAFETY FUNCTION:	These Drywell Sump inboard isolation valves close to provide containment isolation. They remain open to allow pump out of the sumps for drywell leakage monitoring.
Deferred Testing:	ISTC-3510, 5121: Quarterly exercise and stroke time to the closed position.
JUSTIFICATION:	Although these valves are capable of being tested at power, operating experience at other Exelon and non-Exelon plants has demonstrated a potential for valves to fail or degrade as a result of cycling at power (Reference ICES #314926, #314808 and #305581). Failure modes have included severe packing leakage and a loss of containment isolation function. Due to the fact that these valves are located inside the drywell and are "inaccessible", the inability of the valves to open or close would result in a degraded system with the potential for an unnecessary plant shutdown and challenge to safety systems. Additionally, the containment would require de-inerting in order to perform repairs. Thus the risks associated with testing these valves on line vs during refuel outages* outweigh the benefits of testing quarterly on line. During refuel outages, the primary containment is open and any repairs could be performed immediately without cycling the plant and challenging safety systems during the process of shutting down the plant. Thus this analysis provides the basis, in accordance with ISTC-3521(e) for deferring testing from a quarterly test frequency to refuel outages as noted in NUREG 1482 section 2.4.5 "Deferring Valve Testing to Cold Shutdown or Refueling Outages". * It has been determined under this evaluation that testing during cold shutdowns (ref ISTC-3521(c) is not practicable because primary containment (drywell and suppression chambers) is not always made accessible.
REFERENCES:	NUREG-1482 Revision 2, Paragraph 3.1.1 Deferring Valve Testing to Each Cold Shutdown or Refueling Outage NUREG-1482 Revision 2, Paragraph 3.1.1.3 De-inerting Containment if Boiling-Water Reactors to Allow Cold Shutdown Testing

ROJ-35

SYSTEM: **Main Steam**

COMPONENT: 29MOV-74 MST Inboard Line Drain Inboard Isolation Valve

CATEGORY: A

TEST REQUIREMENTS: Quarterly exercise & stroke time testing.

SAFETY FUNCTION: This main steam line drain valve closes to provide containment isolation.

JUSTIFICATION: Exercising this valve during normal operations or cold shutdown would require a shutdown if maintenance was required. In addition, entry into the containment may be prohibited if the drywell remains inerted.

This valve would cause a loss of system function if it was to fail closed during a quarterly test. A unit shutdown would be required to perform corrective maintenance.

Although this valve is capable of being tested at power, operating experience at other Exelon and non-Exelon plants has demonstrated a potential for valves to fail or degrade as a result of cycling at power (Reference ICES #314926, #314808 and #305581). Failure modes have included severe packing leakage and a loss of containment isolation function. Due to the fact that this valve are located inside the drywell and are "inaccessible", the inability of the valve to open or close would result in a degraded system with the potential for an unnecessary plant shutdown and challenge to safety systems. Additionally, the containment would require de-inerting in order to perform repairs.

Thus the risks associated with testing this valve on line vs during refuel outages* outweigh the benefits of testing quarterly on line. During refuel outages, the primary containment is open and any repairs could be performed immediately without cycling the plant and challenging safety systems during the process of shutting down the plant. Thus this analysis provides the basis, in accordance with ISTC-3521(e) for deferring testing from a quarterly test frequency to refuel outages as noted in NUREG 1482 section 2.4.5 "Deferring Valve Testing to Cold Shutdown or Refueling Outages".

* It has been determined under this evaluation that testing during cold shutdowns (ref ISTC-3521(c) is not practicable because primary containment (drywell and suppression chambers) is not always made accessible.

REFERENCES: NUREG-1482 Revision 2, Paragraph 3.1.1 Deferring Valve Testing to Each Cold Shutdown or Refueling Outage

NUREG-1482 Revision 2, Paragraph 3.1.1.3 De-inerting Containment if Boiling-Water Reactors to Allow Cold Shutdown Testing

ATTACHMENT 12

TECHNICAL POSITION INDEX

TECHNICAL POSITION NUMBER	REV #	TITLE

ATTACHMENT 13

TECHNICAL POSITIONS

None

ATTACHMENT 14

INSERVICE TESTING PUMP TABLE

PUMP ID	Description	CLASS	GROUP	DWG No. CO-ORD	PUMP TYPE	DRIVER	TEST	TEST FREQUENCY	RELIEF REQUEST
10P-1A	RHR Service Water Pump A	3	A	FM-20B B-6	VLS	M	ΔP, V_v	Quarterly	PRR-04
							ΔP, V_v	2YR Comp Pump Test	PRR-02 & 04
10P-1B	RHR Service Water Pump B	3	A	FM-20B B-5	VLS	M	ΔP, V_v	Quarterly	PRR-04
							ΔP, V_v	2YR Comp Pump Test	PRR-02 & 04
10P-1C	RHR Service Water Pump C	3	A	FM-20B C-6	VLS	M	ΔP, V_v	Quarterly	PRR-04
							ΔP, V_v	2YR Comp Pump Test	PRR-02 &04
10P-1D	RHR Service Water Pump D	3	A	FM-20B C-5	VLS	M	ΔP, V_v	Quarterly	PRR-04
							ΔP, V_v	2YR Comp Pump Test	PRR-02 & 04
10P-3A	Residual Heat Removal Pump A	2	A	FM-20A C-7	C	M	ΔP, V_v	Quarterly	PRR-03 & 04
							ΔP, V_v	2YR Comp Pump Test	PRR-03 & 04
10P-3B	Residual Heat Removal Pump B	2	A	FM-20A C-4	C	M	ΔP, V_v	Quarterly	PRR-03 & 04
							ΔP, V_v	2YR Comp Pump Test	PRR-03 & 04
10P-3C	Residual Heat Removal Pump C	2	A	FM-20A C-7	C	M	ΔP, V_v	Quarterly	PRR-03 & 04
							ΔP, V_v	2YR Comp Pump Test	PRR-03 & 04
10P-3D	Residual Heat Removal Pump D	2	A	FM-20A C-4	C	M	ΔP, V_v	Quarterly	PRR-03 & 04
							ΔP, V_v	2YR Comp Pump Test	PRR-03 & 04
11P-2A	Standby Liquid Control A Pump	2	B	FM-21A D-4	PDR	M	Q	Quarterly	PRR-04
							Q, V_v	2YR Comp Pump Test	PRR-04
11P-2B	Standby Liquid Control B Pump	2	B	FM-21A B-4	PDR	M	Q	Quarterly	PRR-04
							Q, V_v	2YR Comp Pump Test	PRR-04
14P-1A	Core Spray Pump A	2	B	FM-23A C-8	C	M	ΔP	Quarterly	PRR-01 & 04
							ΔP, V_v	2YR Comp Pump Test	PRR-04
14P-1B	Core Spray Pump B	2	B	FM-23A C-3	C	M	ΔP	Quarterly	PRR-01 & 04
							ΔP, V_v	2YR Comp Pump Test	PRR-04
23P-1B	High Pressure Coolant Injection Booster Pump	2	B	FM-25A E-5	C	T	ΔP	Quarterly	PRR-04
							ΔP, V_v	2YR Comp Pump Test	PRR-04
23P-1M	High Pressure Coolant Injection Main Pump	2	B	FM-25A E-4	C	T	ΔP,,N	Quarterly	PRR-04
							ΔP,N, V_v	2YR Comp Pump Test	PRR-04
46P-2A	Emergency Service Water Pump A	3	B	FM-46B D-8	VLS	M	ΔP	Quarterly	PRR-04
							ΔP, V_v	2YR Comp Pump Test	PRR-04
46P-2B	Emergency Service Water Pump B	3	B	FM-46B C-8	VLS	M	ΔP	Quarterly	PRR-04
							ΔP, V_v	2YR Comp Pump Test	PRR-04

ATTACHMENT 15

INSERVICE TESTING VALVE TABLE

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
01-125MOV-100A	SGT A Decay Heat Cooling Inlet Isolation valve	BTF	MO	FM-48A	C-6	2A	O/C O	B	A	STO STC PI	Q Q Y2			Augmented
01-125MOV-100B	SGT B Decay Heat Cooling Inlet Isolation Valve	BTF	MO	FM-48A	F-6	2A	0/C O	B	A	STO STC PI	Q Q Y2			Augmented
01-125MOV-11	SGT RX BLDG suction above 369 EL Isolation Valve	BTF	MO	FM-48A	G-8	2A	O C	B	A	STO PI	Q Y2			Augmented
01-125MOV-12	SGT RX BLDG suction below 369 EL Isolation Valve	BTF	MO	FM-48A	F-8	2A	O C	B	A	STO PI	Q Y2			Augmented
01-125MOV-14A	SGT Filter Train A Inlet Isolation Valve	BTF	MO	FM-48A	D-6	2A	O/C C	B	A	STO PI	Q Y2			Augmented
01-125MOV-14B	SGT RX BLDG suction above 369 EL Isolation Valve	BTF	MO	FM-48A	E-6	2A	0/C C	B	A	STO PI	Q Y2			Augmented
01-125MOV-15A	SGT Fan A Discharge Isolation Valve	BTF	MO	FM-48A	D-3	2A	O C	B	A	STO PI	Q Y2			Augmented
01-125MOV-15B	SGT Fan B Discharge Isolation Valve	BTF	MO	FM-48A	F-3	2A	O C	B	A	STO PI	Q Y2			Augmented
02AOV-17	ADS Reactor Head Vent Inboard	GL	AO	FM-29A	G-7	1	C C	B	P	PI	Y2			
02AOV-18	ADS Reactor Head Vent Outboard	GL	AO	FM-29A	G-7	1	C C	B	P	PI	Y2			
02RV-1	ADS RV-71A Discharge Vacuum Relief	CK	SA	FM-29A	D-7	2	C O	C	A	SC/SO	RR		ROJ-04	
02RV-2	ADS RV-71B Discharge Vacuum Relief	CK	SA	FM-29A	D-7	2	C O	C	A	SC/SO	RR		ROJ-04	
02RV-3	ADS RV-71C Discharge Vacuum Relief	CK	SA	FM-29A	D-7	2	C O	C	A	SC/SO	RR		ROJ-04	
02RV-4	ADS RV-71D Discharge Vacuum Relief	CK	SA	FM-29A	D-7	2	C O	C	A	SC/SO	RR		ROJ-04	
02RV-5	ADS RV-71E Discharge Vacuum Relief	CK	SA	FM-29A	D-7	2	C O	C	A	SC/SO	RR		ROJ-04	
02RV-6	ADS RV-71F Discharge Vacuum Relief	CK	SA	FM-29A	D-7	2	C O	C	A	SC/SO	RR		ROJ-04	
02RV-7	ADS RV-71G Discharge Vacuum Relief	CK	SA	FM-29A	D-7	2	C O	C	A	SC/SO	RR		ROJ-04	
02RV-8	ADS RV-71H Discharge Vacuum Relief	CK	SA	FM-29A	D-7	2	C O	C	A	SC/SO	RR		ROJ-04	
02RV-9	ADS RV-71J Discharge Vacuum Relief	CK	SA	FM-29A	D-7	2	C O	C	A	SC/SO	RR		ROJ-04	
02RV-10	ADS RV-71K Discharge Vacuum Relief	CK	SA	FM-29A	D-7	2	C O	C	A	SC/SO	RR		ROJ-04	
02RV-11	ADS RV-71L Discharge Vacuum Relief	CK	SA	FM-29A	D-7	2	C O	C	A	SC/SO	RR		ROJ-04	

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
02RV-71A	ADS Main Steam Line A Safety/Relief Valve	RV	SA, AO	FM-29A	G-6	1	O/C C	B/C	A	ETC ETO RLF	RR RR 6 YR	VRR-03		SOV only SOV only
02RV-71B	ADS Main Steam Line A Safety Relief Valve	RV	SA, AO	FM-29A	G-6	1	O/C C	B/C	A	ETC ETO RLF	RR RR 6 YR	VRR-03		SOV only SOV only
02RV-71C	ADS Main Steam Line B Safety Relief Valve	RV	SA, AO	FM-29A	G-6	1	O/C C	B/C	A	ETC ETO RLF	RR RR 6 YR	VRR-03		SOV only SOV only
02RV-71D	ADS Main Steam Line C Safety Relief Valve	RV	SA, AO	FM-29A	F-6	1	O/C C	B/C	A	ETC ETO RLF	RR RR 6 YR	VRR-03		SOV only SOV only
02RV-71E	ADS Main Steam Line C Safety Relief	RV	SA, AO	FM-29A	F-7	1	O/C C	B/C	A	ETC ETO RLF	RR RR 6 YR	VRR-03		SOV only SOV only
02RV-71F	Main Steam Line C Safety Relief Valve	RV	SA, AO	FM-29A	F-7	1	O/C C	B/C	A	ETC ETO RLF	RR RR 6 YR	VRR-03		SOV only SOV only
02RV-71G	ADS Main Steam Line D Safety Relief Valve	RV	SA, AO	FM-29A	F-7	1	O/C C	B/C	A	ETC ETO RLF	RR RR 6 YR	VRR-03		SOV only SOV only
02RV-71H	ADS Main Steam Line D Safety Relief	RV	SA, AO	FM-29A	G-7	1	O/C C	B/C	A	ETC ETO RLF	RR RR 6 YR	VRR-03		SOV only SOV only
02RV-71J	Main Steam Line A Safty/Relief	RV	SA, AO	FM-29A	G-7	1	O/C C	B/C	A	ETC ETO RLF	RR RR 6 YR	VRR-03		SOV only SOV only
02RV-71K	Main Steam Line A Safety/Relief	RV	SA, AO	FM-29A	G-6	1	O/C C	B/C	A	ETC ETO RLF	RR RR 6 YR	VRR-03		SOV only SOV only
02RV-71L	Main Steam Line D Safety Relief Valve	RV	SA, AO	FM-29A	F-7	1	O/C C	B/C	A	ETC ETO RLF	RR RR 6 YR	VRR-03		SOV only SOV only
02VB-1	ADS RV-71A Discharge Vacuum Breaker	CK	SA	FM-29A	C-7	2	O/C C	C	A	SC/SO	RR		ROJ-4	
02VB-2	ADS RV-71B Discharge Vacuum Breaker	CK	SA	FM-29A	C-7	2	O/C C	C	A	SC/SO	RR		ROJ-4	
02VB-3	ADS RV-71C Discharge Vacuum Breaker	CK	SA	FM-29A	C-7	2	O/C C	C	A	SC/SO	RR		ROJ-4	
02VB-4	ADS RV-71D Discharge Vacuum Breaker	CK	SA	FM-29A	C-7	2	O/C C	C	A	SC/SO	RR		ROJ-4	
02VB-5	ADS RV-71E Discharge Vacuum Breaker	CK	SA	FM-29A	C-7	2	O/C C	C	A	SC/SO	RR		ROJ-4	
02VB-6	ADS RV-71F Discharge Vacuum Breaker	CK	SA	FM-29A	C-7	2	O/C C	C	A	SC/SO	RR		ROJ-4	

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
02VB-7	ADS RV-71G Discharge Vacuum Breaker	CK	SA	FM-29A	C-7	2	O/C C	C	A	SC/SO	RR		ROJ-4	
02VB-8	ADS RV-71H Discharge Vacuum Breaker	CK	SA	FM-29A	C-7	2	O/C C	C	A	SC/SO	RR		ROJ-4	
02VB-9	ADS RV-71J Discharge Vacuum Breaker	CK	SA	FM-29A	C-7	2	O/C C	C	A	SC/SO	RR		ROJ-4	
02VB-10	ADS RV-71K Discharge Vacuum Breaker	CK	SA	FM-29A	C-7	2	O/C C	C	A	SC/SO	RR		ROJ-4	
02VB-11	ADS RV-71L Discharge Vacuum Breaker	CK	SA	FM-29A	C-7	2	O/C C	C	A	SC/SO	RR		ROJ-4	
02-2AOV-39	Recirc Loop Inboard Sample Isolation Valve	GA	AO	FM-26A	E-4	1	C O	A	A	FC LTJ PI STC	RR M30 Y2 RR		ROJ-33 ROJ-33	
02-2AOV-40	Recirc Loop Outboard Sample Isolation Valve	GA	AO	FM-26A	E-3	1	C O	A	A	FC LTJ PI STC	Q M30 Y2 Q			
02-2EFV1-DPT-111A	Recirc Pump A 02DPT-111A Lo Side X-32BC Excess Flow Check Valve	XFC	SA	FM-26A	E-3	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV1-DPT-111B	Recirc Pump B 02DPT-111B Lo Side X-32AA Excess Flow Check Valve	XFC	SA	FM-26A	E-8	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV1-FT-110A	Recirc Loop B 02FT-110A&B Lo Side X-31BB Excess Flow Check Valve	XFC	SA	FM-26A	F-3	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV1-FT-110C	Recirc Loop A 02FT-110C&B Lo Side X-31BB Excess Flow Check Valve	XFC	SA	FM-26A	D-3	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV1-FT-110E	Recirc Loop B 02FT-110E&F Hi Side X-32AC Excess Flow Check Valve	XFC	SA	FM-26A	F-8	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV1-FT-110G	Recic Loop B 02FT-110G&H Lo Side X-31AB Excess Flow Check Valve	XFC	SA	FM-26A	D-8	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV2-DPT-111A	Recirc Pump ADS Logic 02DPT-111A Hi Side X-32BD Excess Flow Check Valve	XFC	SA	FM-26A	E-3	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV2-DPT-111B	Recirc Pump B ADS Logic 02DPT-111B Hi Side X-32AD Excess Flow Check Valve	XFC	SA	FM-26A	E-8	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV2-FT-110A	Recirc Loop A ADS Logic 02ft-110A & E Hi Side X-32AE Excess Flow Check Valve	XFC	SA	FM-26A	F-3	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
02-2EFV2-FT-110C	Recirc Loop A ADS Logic 02ft-110C & D Hi Side X-32BB Excess Flow Check Valve	XFC	SA	FM-26A	D-3	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV2-FT-110E	Recirc Loop B ADS Logic 02FT-110E&F Lo Side X-32AB Excess Flow Check Valve	XFC	SA	FM-26A	F-8	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV2-FT-110G	Recirc Loop B ADS Logic 02FT-110G&H Lo Side X-28BC Excell Flow Check Valve	XFC	SA	FM-26A	D-8	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV-PS-128A	RWR Loop A ADS Logic 02PS-128A X-31BA Excess Flow Check Valve	XFC	SA	FM-26A	B-6	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV-PS-128B	RWR Loop B ADS Logic 02PS-128B X-31AA Excess Flow Check Valve	XFC	SA	FM-26A	B-6	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV-24A	Recirc Pump A Seal Cavity 2 Pressure Instruments X-56D Excess Flow Check Valve	XFC	SA	FM-26A	C-3	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV-24B	Recirc Pump B Seal Cavity 2 Pressure Instruments X-40EB Excess Flow Check Valve	XFC	SA	FM-26A	C-8	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV-25A	Recirc Pump A Seal Cavity 1 Pressure Instruments X-56A Excess Flow Check Valve	XFC	SA	FM-26A	C-3	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2EFV-25B	Recirc Pump B Seal Cavity 1 Pressure Instruments X-40EA Excess Flow Check Valve	XFC	SA	FM-26A	C-8	1	C O	A/C	A	CC LT	RR	VRR-02 VRR-02		
02-2MOV-53A	Reactor Water Recirc Pump A Discharge Isolation Valve	GA	SA	FM-26A	C-3	1	C O	B	A	PI STC	RR CS		 CSJ-01	
02-2MOV-53B	Reactor Water Recirc Pump B Discharge Isolation Valve	GA	SA	FM-26A	C-8	1	C O	B	A	PI STC	RR CS		 CSJ-01	
02-2RWR-13A	RWR Pump A Seal Purge Supply Check Valve	CK	SA	FM-26A	C-3	1	C O	A/C	A	CCF COF LTJ	Q RR AJ		 ROJ-02	
02-2RWR-13B	RWR Pump B Seal Purge Supply Check Valve	CK	SA	FM-26A	C-8	1	C O	A/C	A	CCF COF LTJ	Q RR AJ		 ROJ-02	
02-2RWR-41A	Recirc Pump A Seal Purge Check Valve	CK	SA	FM-26A	D-3	1	C O	A/C	A	CCF COF LTJ	Q RR AJ		 ROJ-03	
02-2RWR-41B	Recirc Pump A Seal Purge Check Valve	CK	SA	FM-26A	D-8	1	C O	A/C	A	CCF COF LTJ	Q RR AJ		 ROJ-03	
02-3EFV-11	Reactor Vessel Instrumentation Excess Flow	XFC	SA	FM-47A	F-7	1	C O	A/C	A	CC LT	RR RR	VRR-02		

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
02-3EFV-13A	Reactor Vessel Instrumentation Excess Flow	XFC	SA	FM-47A	E-7	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-13B	Reactor Vessel Instrumentation Excess Flow	XFC	SA	FM-47A	E-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-15A	Reactor Vessel Instrumentation Excess Flow	XFC	SA	FM-47A	E-7	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-15B	Reactor Vessel Instrumentation Excess Flow	XFC	SA	FM-47A	E-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-15N	Reactor Vessel Instrumentation Excess Flow	XFC	SA	FM-47A	B-7	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-17A	Reactor Vessel Instrumentation Excess Flow	XFC	SA	FM-47A	D-7	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-17B	Reactor Vessel Instrumentation Excess Flow	XFC	SA	FM-47A	D-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-19A	Reactor Vessel Instrumentation Excess Flow	XFC	SA	FM-47A	D-7	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-19B	Reactor Vessel Instrumentation Excess Flow	XFC	SA	FM-47A	D-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-21A	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	H-5	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-21B	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	C-7	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-21C	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	C-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-21D	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	H-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-23	Reactor Vessel Instrumentation Excess Flow	XFC	SA	FM-47A	F-7	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-23A	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	H-5	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-23B	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	D-7	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-23C	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	D-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-23D	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	C-7	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-25	CRD Pressure Sensing Instrumentation Excess Flow	XFC	SA	FM-47A	C-7	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31A	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	H-5	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31B	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	H-5	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31C	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	H-5	1	C O	A/C	A	CC LT	RR RR	VRR-02		

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
02-3EFV-31D	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	H-5	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31E	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	D-7	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31F	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	H-5	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31G	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	G-5	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31H	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	G-5	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31J	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	H-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31K	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	H-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31L	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	H-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31M	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	D-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31N	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	H-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31P	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	H-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31R	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	G-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-31S	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	G-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
02-3EFV-33	Jet Pump Instrumentation Excess Flow	XFC	SA	FM-47A	B-4	1	C O	A/C	A	CC LT	RR RR	VRR-02		
03AOV-126	HCU Inlet Scram (Typical of 137)	GL	AO	FM-27B	C-4	2	O C	B	A					Exempt-Skid
03AOV-127	HCU Outlet Scram (Typical of 137)	GL	AO	FM-27B	D-4	2	O C	B	A					Exempt-Skid
03AOV-32	Scram Discharge Volume B Vent	GL	AO	FM-27B	H-4	2	C O	B	A	FC STC PI	Q Q Y2			
03AOV-33	Scram Discharge Volume B Drain	GL	AO	FM-27B	F-4	2	C O	B	A	FC STC PI	Q Q Y2			
03AOV-34	Scram Discharge Volume B Vent	GL	AO	FM-27B	H-4	2	C O	B	A	FC STC PI	Q Q Y2			
03AOV-35	Scram Discharge Volume B Drain	GL	AO	FM-27B	F-4	2	C O	B	A	FC STC PI	Q Q Y2			

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
03AOV-36	Scram Discharge Volume A Vent	GL	AO	FM-27B	H-6	2	C O	B	A	FC STC PI	Q Q Y2			
03AOV-37	Scram Discharge Volume A Drain	GL	AO	FM-27B	F-6	2	C O	B	A	FC STC PI	Q Q Y2			
03AOV-38	Scram Discharge Volume A Vent	GL	AO	FM-27B	H-6	2	C O	B	A	FC STC PI	Q Q Y2			
03AOV-39	Scram Discharge Volume A Drain	GL	AO	FM-27B	F-6	2	C O	B	A	FC STC PI	Q Q Y2			
03HCU-114	Scram Discharge Line Check (Typical of 137)	XFC	SA	FM-27B	D-4	2	O O	C	A					Exempt Skid
03HCU-115	Charging Water Inlet (Typical of 137)	XFC	SA	FM-27B	C-4	2	C C	C	A				ROJ-27	Exempt Skid
03HCU-138	Cooling Water Supply (Typical of 137)	XFC	SA	FM-27B	C-4	2	C C	C	A					Exempt Skid
03SOV-120	Withdraw Settle Solenoid	GA	AO	FM-27B	C-4	2	C O	B	A					Exempt Skid
03SOV-121	Insert Exhaust Solenoid	GA	AO	FM-27B	C-4	2	C O	B	A					Exempt Skid
03SOV-122	Withdraw Drive Water Solenoid	GA	AO	FM-27B	C-4	2	C O	B	A					Exempt Skid
03SOV-123	Withdraw Settle Solenoid	GA	AO	FM-27B	C-4	2	C O	B	A					Exempt Skid
07EV-104A	TIP A Explosive Shear Valve	SHR	EXP	FM-119A	F-5	2A	C O	D	A	DT	Y2			Augmented
07EV-104B	TIP B Explosive Shear Valve	SHR	EXP	FM-119A	F-4	2A	C O	D	A	DT	Y2			Augmented
07EV-104C	TIP C Explosive Shear Valve	SHR	EXP	FM-119A	F-4	2A	C O	D	A	DT	Y2			Augmented
07SOV-104A	TIP A Ball Valve	BAL	SO	FM-119A	F-5	2A	C O/C	A	A	FC LTJ PI STC	Q M60 Y2 Q	VRR-01		
07SOV-104B	TIP B Ball Valve	BAL	SO	FM-119A	F-4	2A	C O/C	A	A	FC LTJ PI STC	Q M60 Y2 Q	VRR-01		
07SOV-104C	TIP C Ball Valve	BAL	SO	FM-119A	F-4	2A	C O/C	A	A	FC LTJ PI STC	Q M60 Y2 Q	VRR-01		

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
10AOV-68A	RHR A LPCI Testable Check Valve	CK	SA, AO	FM-20A	F-6	1	O/C C	A/C	A	COF CCF LTH	RR RR Y2	VRR-04	ROJ-29 ROJ-29	
10AOV-68B	RHR B LPCI Testable Check Valve	CK	SA, AO	FM-20A	F-5	1	O/C O	A/C	A	COF CCF LTH	RR RR Y2	VRR-04	ROJ-29 ROJ-29	
10AOV-71A	RHR Heat Exchanger A Outlet to Torus or RCIC Isol Valve	GL	AO	FM-20B	F-6	2	C C	B	A	PI	Y2			
10AOV-71B	RHR Heat Exchanger B Outlet to Torus or RCIC Isol Valve	GL	AO	FM-20B	F-5	2	C C	B	A	PI	Y2			
10MOV-12A	RHR Heat Exchanger A Outlet Isolation Valve	GA	MO	FM-20B	B-6	2	O O	B	P	PI	Y2			
10MOV-12B	RHR Heat Exchanger B Outlet Isolation Valve	GA	MO	FM-20B	F-5	2	O O	B	P	PI	Y2			
10MOV-13A	RHR Pump A Suct Torus Isolation valve	GA	MO	FM-20A	B-6	2	O/C O	B	A	PI STC STO	Y2 Q Q			
10MOV-13B	RHR Pump B Suct Torus Isolation valve	GA	MO	FM-20A	C-4	2	O/C O	B	A	PI STC STO	Y2 Q Q			
10MOV-13C	RHR Pump C Suct Torus Isolation valve	GA	MO	FM-20A	C-6	2	O/C O	B	A	PI STC STO	Y2 Q Q			
10MOV-13D	RHR Pump D Suct Torus Isolation valve	GA	MO	FM-20A	C-5	2	O/C O	B	A	PI STC STO	Y2 Q Q			
10MOV-148A	RHRSW A To RHR Cross Tie Upstream Isolation Valve	GA	MO	FM-20B	E-8	3	C C	B	P	PI	Y2			
10MOV-148B	RHRSW B To RHR Cross Tie Upstream Isolation Valve	GA	MO	FM-20B	E-2	3	C C	B	P	PI	Y2			
10MOV-149A	RHRSW A To RHR Cross Tie Downstream Isolation Valve	GA	MO	FM-20B	D-8	3	C C	B	P	PI	Y2			
10MOV-149B	RHRSW B To RHR Cross Tie Downstream Isolation Valve	GA	MO	FM-20B	D-2	3	C C	B	P	PI	Y2			
10MOV-15A	RHR A Pump Shutdown Cooling Isolation Valve	GA	MO	FM-20A	C-6	2	C O/C	B	A	PI STC	Y2 Q			
10MOV-15B	RHR B Pump Shutdown Cooling Isolation Valve	GA	MO	FM-20A	C-4	2	C O/C	B	A	PI STC	Y2 Q			
10MOV-15C	RHR C Pump Shutdown Cooling Isolation Valve	GA	MO	FM-20A	C-6	2	C O/C	B	A	PI STC	Y2 Q			
10MOV-15D	RHR D Pump Shutdown Cooling Isolation Valve	GA	MO	FM-20A	C-4	2	C O/C	B	A	PI STC	Y2 Q			
10MOV-167A	RHR Heat Exch A DNSTR Vent to Torus Isolation Valve	GL	MO	FM-20B	F-8	2	C C	B	P	PI	Y2			

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
10MOV-167B	RHR Heat Exch A DNSTR Vent to Torus Isolation Valve	GL	MO	FM-20B	F-3	2	C C	B	P	PI	Y2			
10MOV-16A	RHR Loop A Min Flow Isolation Valve	GA	MO	FM-20A	D-8	2	O/C O/C	B	A	PI STC STO	Y2 Q Q			
10MOV-16B	RHR Loop B Min Flow Isolation Valve	GA	MO	FM-20A	D-3	2	O/C O/C	B	A	PI STC STO	Y2 Q Q			
10MOV-17	RHR Shutdown Cooling Outboard Isolation Valve	GA	MO	FM-20A	D-5	1	C C	A	A	LT PI STC	 Y2 RR	VRR-04	 ROJ-28	
10MOV-18	RHR Shutdown Cooling Inboard Isolation Valve	GA	MO	FM-20A	E-5	1	C C	A	A	LT PI STC	 Y2 RR	VRR-04	 ROJ-28	
10MOV-25A	RHR A LPCI Inboard Injection Valve	GA	MO	FM-20A	F-8	1	O/C C	A	A	LT LTJ PI STC STO	 AJ Y2 CS CS	VRR-04	 CSJ-10 CSJ-10	
10MOV-25B	RHR B LPCI Inboard Injection Valve	GA	MO	FM-20A	F-3	1	O/C C	A	A	LT LTJ PI STC STO	 AJ Y2 CS CS	VRR-04	 CSJ-10 CSJ-10	
10MOV-26A	RHR A Containment Spray Outboard Isolation Valve	GA	MO	FM-20A	G-7	2	O/C C	A	A	PI STC STO	Y2 Q Q			
10MOV-26B	RHR B Containment Spray Outboard Isolation Valve	GA	MO	FM-20A	G-4	2	O/C C	A	A	PI STC STO	Y2 Q Q			
10MOV-27A	RHR A LPCI Outboard Injection Valve	AN-GL	MO	FM-20A	F-8	1	O/C O/C	A	A	PI STC STO	Y2 Q Q			Angle Globe Valve
10MOV-27B	RHR B LPCI Outboard Injection Valve	AN-GL	MO	FM-20A	F-3	1	O/C O/C	A	A	PI STC STO	Y2 Q Q			Angle Globe Valve
10MOV-31A	RHR A Containment Spray Inboard Isolation Valve	GL	MO	FM-20A	G-6	2	O/C C	A	A	LT PI STC STO	AJ Y2 Q Q			
10MOV-31B	RHR B Containment Spray Inboard Isolation Valve	GL	MO	FM-20A	G-5	2	O/C C	A	A	LT PI STC STO	AJ Y2 Q Q			

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
10MOV-34A	RHR A Torus Cooling Supply Valve	GL	MO	FM-20A	E-7	2	O/C C	B	A	PI STC STO	Y2 Q Q			
10MOV-34B	RHR B Torus Cooling Supply Valve	GL	MO	FM-20A	E-3	2	O/C C	B	A	PI STC STO	Y2 Q Q			
10MOV-38A	RHR A To Torus Spray Isolation Valve	GL	MO	FM-20A	E-7	2	O/C C	A	A	LTJ PI STC STO	AJ Y2 Q Q			
10MOV-38B	RHR B To Torus Spray Isolation Valve	GL	MO	FM-20A	E-4	2	O/C C	A	A	LTJ PI STC STO	AJ Y2 Q Q			
10MOV-39A	RHR A Torus Cooling Isolation Valve	GL	MO	FM-20A	E-8	2	O/C C	A	A	PI STC STO	Y2 Q Q			
10MOV-39B	RHR Loop B Torus Cooling Isolation Valve	GL	MO	FM-20A	E-3	2	O/C C	A	A	PI STC STO	Y2 Q Q			
10MOV-65A	RHR Heat Exchanger A Shell Inlet Isolation Valve	GL	MO	FM-20B	G-6	2	O O	B	P	PI	Y2			
10MOV-65B	RHR Heat Exchanger B Shell Inlet Isolation Valve	GL	MO	FM-20B	G-5	2	O O	B	P	PI	Y2			
10MOV-66A	RHR Heat Exchanger A Bypass valve	GL	MO	FM-20A	D-8	2	O/C O	B	A	PI STC STO	Y2 Q Q			
10MOV-66B	RHR Heat Exchanger B Bypass valve	GL	MO	FM-20A	D-3	2	O/C O	B	A	PI STC STO	Y2 Q Q			
10MOV-89A	RHR Heat Exchanger A Service Water Outlet Isolation Valve	GA	MO	FM-20B	D-6	3	O C	B	A	PI STC STO	Y2 Q Q			
10MOV-89B	RHR Heat Exchanger B Service Water Outlet Isolation Valve	GA	MO	FM-20B	E-5	3	O C	B	A	PI STC STO	Y2 Q Q			
10RHR-14A	RHRSW Pump A Discharge Check Valve	CK	SA	FM-20B	B-7	3	O/C C	C	A	DIS CCF COF	RR Q Q			CVCM JAF-03
10RHR-14B	RHRSW Pump B Discharge Check Valve	CK	SA	FM-20B	B-4	3	O/C C	C	A	DIS CCF COF	RR Q Q			CVCM JAF-03
10RHR-14C	RHRSW Pump C Discharge Check Valve	CK	SA	FM-20B	C-7	3	O/C C	C	A	DIS CCF COF	RR Q Q			CVCM JAF-03

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
10RHR-14D	RHRSW Pump D Discharge Check Valve	CK	SA	FM-20B	C-4	3	O/C C	C	A	DIS CCF COF	RR Q Q			CVCM JAF-03
10RHR-262	RHR Loop B Reactor Head Spray Keep Full Inner Check Valve	CK	SA	FM-20A	H-3	2	C O/C	C	A	LT COF				
10RHR-277	RHR Loop A Reactor Head Spray Keep Full Inner Check Valve	CK	SA	FM-20A	G-8	2	C O/C	C	A	LT COF				
10RHR-42A	RHR Pump A Discharge Check Valve	CK	SA	FM-20A	C-8	2	O/C O/C	C	A	DIS CCF COF	RR Q Q			
10RHR-42B	RHR Pump B Discharge Check Valve	CK	SA	FM-20A	C-3	2	O/C O/C	C	A	DIS CCF COF	RR Q Q			
10RHR-42C	RHR Pump C Discharge Check Valve	CK	SA	FM-20A	C-8	2	O/C O/C	C	A	DIS CCF COF	RR Q Q			
10RHR-42D	RHR Pump D Discharge Check Valve	CK	SA	FM-20A	C-3	2	O/C O/C	C	A	DIS CCF COF	RR Q Q			
10RHR-64A	RHR Pump A Min Flow Check Valve	CK	SA	FM-20A	C-8	2	O/C O/C	C	A	DIS CCF COF	RR Q Q		ROJ-05 ROJ-05	CVCM JAF-02
10RHR-64B	RHR Pump B Min Flow Check Valve	CK	SA	FM-20A	C-3	2	O/C O/C	C	A	DIS CCF COF	RR Q Q		ROJ-05 ROJ-05	CVCM JAF-02
10RHR-64C	RHR Pump C Min Flow Check Valve	CK	SA	FM-20A	D-8	2	O/C O/C	C	A	DIS CCF COF	RR Q Q		ROJ-05 ROJ-05	CVCM JAF-02
10RHR-64D	RHR Pump D Min Flow Check Valve	CK	SA	FM-20A	D-3	2	O/C O/C	C	A	DIS CCF COF	RR Q Q		ROJ-05 ROJ-05	CVCM JAF-02
10RHR-95A	RHR Keep Full Pump A Min Flow Check Valve	CK	SA	FM-20A	D-8	2	C O/C	C	A	CCF COF	RR Q		ROJ-06	
10RHR-95B	RHR Keep Full Pump B Min Flow Check Valve	CK	SA	FM-20A	B-5	2	C O/C	C	A	CCF COF	RR Q		ROJ-06	
10RV-41A	RHR Pump A Suction Relief Valve	RV	SA	FM-20A	C-7	2	C C	C	A	RT	Y10			Thermal
10RV-41B	RHR Pump B Suction Relief Valve	RV	SA	FM-20A	C-4	2	C C	C	A	RT	Y10			Thermal
10RV-41C	RHR Pump C Suction Relief Valve	RV	SA	FM-20A	C-7	2	C C	C	A	RT	Y10			Thermal
10RV-41D	RHR Pump D Suction Relief Valve	RV	SA	FM-20A	C-4	2	C C	C	A	RT	Y10			Thermal

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
10RV-43A	RHR Heat Exchanger A Tube Side Relief Valve	RV	SA	FM-20B	E-7	3	O C	C	A	RT	Y10			Thermal
10RV-43B	RHR Heat Exchanger B Tube Side Relief Valve	RV	SA	FM-20B	E-4	3	O C	C	A	RT	Y10			Thermal
10RV-46A	RHR Heat Exchanger A Shell Side Relief Valve	RV	SA	FM-20B	F-7	2	O C	C	A	RT	Y10			Thermal
10RV-46B	RHR Heat Exchanger B Shell Side Relief Valve	RV	SA	FM-20B	F-3	2	O C	C	A	RT	Y10			Thermal
10SOV-203	RHR Loop A Sample Solenoid Valve	GA	SO	FM-18C	E-7	2	C C	B	P	PI	Y2			
10SOV-204	RHR Loop B Sample Solenoid Valve	GA	SO	FM-18C	D-7	2	C C	B	P	PI	Y2			
10SV-35A	RHR Loop A Safety Valve	RV	SA	FM-20A	E-8	2	C C	C	A	RT	Y10			Y4 due to past performance
10SV-35B	RHR Loop B Safety Valve	RV	SA	FM-20A	E-3	2	C C	C	A	RT	Y10			Y4 due to past performance
10SV-40	RHR Shutdown Cooling Relief Valve	RV	SA	FM-20A	D-5	2	C C	C	A	RT	Y10			Y4 Group of one
11EV-14A	SLC A Double Squib Activated Shear Explosive Valve	SHR	EXP	FM-21A	D-6	1	O C	D	A	DT	Y2			
11EV-14B	SLC B Double Squib Activated Shear Explosive Valve	SHR	EXP	FM-21A	B-6	1	O C	D	A	DT	Y2			
11SLC-16	SLC Pumps Discharge to RX Outboard Check Valve	CK	SA	FM-21A	C-7	1	O/C O/C	A/C	A	COF CCF	RR RR		ROJ-07 ROJ-07	
11SLC-17	SLC Pumps Discharge to RX Inboard Check Valve	CK	SA	FM-21A	D-7	1	O/C O/C	A/C	A	COF CCF	RR RR		ROJ-07 ROJ-07	
11SLC-18	SLC Pumps Discharge to RX Inboard Check Valve	GL	M	FM-21A	D-7	1	O O	B	P	PI	Y2			
11SLC-43A	SLC Pump A Discharge Check Valve	CK	SA	FM-21A	D-6	2	O O/C	C	A	DIS CCF COF	CM Q Q			CVCM JAF-04
11SLC-43B	SLC Pump B Discharge Check Valve	CK	SA	FM-21A	B-6	2	O O/C	C	A	DIS CCF COF	CM Q Q			CVCM JAF-04
11SV-39A	SLC Pump 2A Discharge Safety Valve	RV	SA	FM-21A	D-4	2	C C	C	A	RT	Y10			Y8 Group of two
11SV-39B	SLC Pump 2B Discharge Safety Valve	RV	SA	FM-21A	C-4	2	C C	C	A	RT	Y10			Y8 Group of two
12MOV-15	RWCU Supply Inboard Isolation Valve	GA	MO	FM-24A	E-8	1	C O	A	A	LTJ PI STC	AJ Y2 CS		 CSJ-07	
12MOV-18	RWCU Supply Outboard Isolation Valve	GA	MO	FM-24A	E-7	1	C O	A	A	LTJ PI STC	AJ Y2 CS		 CSJ-07	

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
12MOV-69	RWCU Return Containment Isolation Valve	GA	MO	FM-24A	H-7	1	C O	A	A	LTJ PI STC	AJ Y2 CS		 CSJ-07	
13EFV-01A	RCIC Turbine Steam Supply Leak Detect Instruments Excess Flow Check Valve	CK	SA	FM-22A	G-7	1	C O	A/C	A	CC LK	RR RR	VRR-02 VRR-02		
13EFV-01B	RCIC Turbine Steam Supply Leak Detect Instruments Excess Flow Check Valve	CK	SA	FM-22A	G-7	1	C O	A/C	A	CC LK	RR RR	VRR-02 VRR-02		
13EFV-02A	RCIC Turbine Steam Supply Leak Detect Instruments Excess Flow Check Valve	CK	SA	FM-22A	G-7	1	C O	A/C	A	CC LK	RR RR	VRR-02 VRR-02		
13EFV-02B	RCIC Turbine Steam Supply Leak Detect Instruments Excess Flow Check Valve	CK	SA	FM-22A	F-7	1	C O	A/C	A	CC LK	RR RR	VRR-02 VRR-02		
13MOV-130	RCIC Turbine Exhaust Line Vacuum Breaker Valve	GA	MO	FM-22A	E-6	2	O O	B	P	PI	Y2			
13MOV-15	RCIC Steam Supply Inboard Isolation Valve	GA	MO	FM-22A	F-7	1	C O	A	A	LTJ PI STC	AJ Y2 RR		 ROJ-31	
13MOV-16	RCIC Turbine Steam Supply Outboard Isolation Valve	GA	MO	FM-22A	F-7	1	C O	A	A	LTJ PI STC	AJ Y2 Q			
13MOV-21	RCIC Pump Discharge to Reactor Inboard Valve	GA	MO	FM-22A	F-5	1	C C	A	A	LTJ PI STC	AJ Y2 Q			
13MOV-27	RCIC Pump Min Flow Isolatio Valve	GL	MO	FM-22A	E-5	2	C C	B	A	PI STC	Y2 Q			
13MOV-41	RCIC Pump Suction From Suppression Pool Inboard Isolation Valve	GA	MO	FM-22A	D-7	2	C C	B	A	PI STC	Y2 Q			
13RCIC-22	RCIC Testable Check Valve	CK	SA	FM-22A	F-6	1A	C C	C	A	SO	CS		CSJ-09	Augmented
13RCIC-37	RCIC Turbine Exhaust Vacuum Break Check Valve	CK	SA	FM-22A	E-6	2	O O	C	A	DIS CCF CCO	Y2 RR RR		 ROJ-24 ROJ-24	
13RCIC-38	RCIC Turbine Exhaust Vacuum Break Check Valve	CK	SA	FM-22A	E-6	2	O O	C	A	DIS CCF CCO	Y2 RR RR		 ROJ-24 ROJ-24	
13RCIC-4	RCIC Turbine Exhaust to Tours Downstream Check Valve	CK	SA	FM-22A	D-6	2	O/C O	A/C	A	LTJ CCF COF	AJ RR Q		 ROJ-08	 CVCM JAF-05
13RCIC-5	RCIC Turbine Exhaust to Torus Upstream Check Valve	CK	SA	FM-22A	C-6	2	O/C O	A/C	A	LTJ CCF COF	AJ RR Q		 ROJ-08	 CVCM JAF-05

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
13RCIC-7	RCIC Vacuum Pump P-3 Discharge Stop Check Valve	CK	SA	FM-22A	C-7	2	C O	C	A	CCF COF	RR RR		ROJ-25 ROJ-25	
14AOV-13A	CSP A Reactor Isolation Testable Check Valve	CK	SA, AO	FM-23A	G-6	1	O/C C	A/C	A	LT PI SC/SO	RR Y2 RR	VRR-04	 ROJ-09	 CVCM JAF-06
14AOV-13B	CSP B Reactor Isolation Testable Check Valve	CK	SA, AO	FM-23A	G-5	1	O/C C	A/C	A	LT PI SC/SO	RR Y2 RR	VRR-04	 ROJ-09	 CVCM JAF-06
14CSP-10A	Core Spray Pump A Discharge Check Valve	CK	SA	FM-23A	D-8	2	O C	C	A	CCF COF	Q Q			CVCM JAF-07
14CSP-10B	Core Spray Pump B Discharge Check Valve	CK	SA	FM-23A	D-3	2	O C	C	A	CCF COF	Q Q			CVCM JAF-07
14CSP-62A	Core Spray Hold Pump A Min Flow Check Valve	CK	SA	FM-23A	E-7	2	O/C C	C	A	CCF COF	RR Q		ROJ-10	
14CSP-62B	Core Spray Hold Pump B Min Flow Check Valve	CK	SA	FM-23A	E-3	2	O/C C	C	A	CCF COF	RR Q		ROJ-10	
14CSP-76A	Core Spray Loop A Keep Full Check Valve	CK	SA	FM-23A	F-7	2	C O	C	A	CCF COF	Q Q			
14CSP-76B	Core Spray Loop B Keep Full Check Valve	CK	SA	FM-23A	F-4	2	C O	C	A	CCF COF	Q Q			
14EFV-31A	Core Spray Loop A Spray Nozzle 14DPIS-43A Excess Flow Check Valve	CK	SA	FM-23A	E-4	1	O/C O	A/C	A	LT SC	RR RR	VRR-02 VRR-02		
14EFV-31B	Core Spray Loop B Spray Nozzle 14DPIS-43A Excess Flow Check Valve	CK	SA	FM-23A	E-4	1	O/C O	A/C	A	LT SC	RR RR	VRR-02 VRR-02		
14MOV-11A	Core Spray Loop A Outboard Isolation Valve	GS	MO	FM-23A	F-7	1	O O	A	A	PI STC STO	Y2 Q Q			
14MOV-11B	Core Spray Loop B Outboard Isolation Valve	GS	MO	FM-23A	F-4	1	O O	A	A	PI STC STO	Y2 Q Q			
14MOV-12A	Core Spray Loop A Inboard Isolation Valve	GA	MO	FM-23A	F-6	1	O/C C	A	A	LTJ LT PI STC STO	AJ Y2 CS CS	 VRR-04	 CSJ-11 CSJ-11	
14MOV-12B	Core Spray Loop B Inboard Isolation Valve	GA	MO	FM-23A	F-4	1	O/C C	A	A	LTJ LT PI STC STO	AJ Y2 CS CS	 VRR-04	 CSJ-11 CSJ-11	
14MOV-26A	CSP A Full Flow Test to Suppression Pool Isolation Valve	GL	MO	FM-23A	F-7	2	C C	B	A	PI STC	Y2 Q			

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
14MOV-26B	CSP B Full Flow Test to Suppression Pool Isolation Valve	GL	MO	FM-23A	F-3	2	C C	B	A	PI STC	Y2 Q			
14MOV-5A	CSP Pump A Min Flow Isolation Valve	GA	MO	FM-23A	E-7	2	O/C C	B	A	PI STC STO	Y2 Q Q			
14MOV-5B	CSP Pump B Min Flow Isolation Valve	GA	MO	FM-23A	E-3	2	O/C C	B	A	PI STC STO	Y2 Q Q			
14MOV-7A	CSP Pump A Suction From Suppression Pool Isolation Valve	GA	MO	FM-23A	C-6	2	O/C O	B	A	PI STC STO	Y2 Q Q			
14MOV-7B	CSP Pump B Suction From Suppression Pool Isolation Valve	GA	MO	FM-23A	C-4	2	O/C O	B	A	PI STC STO	Y2 Q Q			
14SV-20A	Core Spray Pump A Discharge Safety Valve	RV	SA	FM-23A	E-8	2	C C	C	A	RT	Y10			Y8 group of two
14SV-20B	Core Spray Pump B Discharge Safety Valve	RV	SA	FM-23A	E-2	2	C C	C	A	RT	Y10			Y8 group of two
15AOV-130A	RBCLC to Drywell Cooling Assembly A Isolation	GL	AO	FM-15B	C-7	2A	C O	A	A	LTJ PI STC	AJ Y2 CS		CSJ-02	Augmented
15AOV-130B	RBCLC to Drywell Cooling Assembly B Isolation	GL	AO	FM-15B	D-4	2A	C O	A	A	LTJ PI STC	AJ Y2 CS		CSJ-02	Augmented
15AOV-131A	RBCLC Return from Drywell Cooling Assembly A Isolation	GL	AO	FM-15B	E-7	2A	C O	A	A	LTJ PI STC	AJ Y2 CS		CSJ-02	Augmented
15AOV-131B	RBCLC Return from Drywell Cooling Assembly B Isolation	GL	AO	FM-15B	E-4	2A	C O	A	A	LTJ PI STC	AJ Y2 CS		CSJ-02	Augmented
15AOV-132A	RBCLC Supply To Recric Pump & Motor A Coolers Isolation	GL	AO	FM-15B	F-4	2A	C O	A	A	LTJ PI STC	AJ Y2 CS		CSJ-02	Augmented
15AOV-132B	RBCLC Supply To Recric Pump & Motor B Coolers Isolation	GL	AO	FM-15B	F-7	2A	C O	A	A	LTJ PI STC	AJ Y2 CS		CSJ-02	Augmented
15AOV-133A	RBCLC Return From Recric Pump & Motor A Coolers Isolation	GL	AO	FM-15B	F-4	2A	C O	A	A	LTJ PI STC	AJ Y2 CS		CSJ-02	Augmented
15AOV-133B	RBCLC Return From Recric Pump & Motor B Coolers Isolation	GL	AO	FM-15B	F-7	2A	C O	A	A	LTJ PI STC	AJ Y2 CS		CSJ-02	Augmented
15AOV-134A	RBCLC supply to Drywell Equipment Sump Cooler Isolation	GL	AO	FM-15B	C-6	2A	C O	A	A	LTJ PI STC	AJ Y2 CS		CSJ-02	Augmented

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
15RBC-214	PASS Cooler Emergency Water Supply Check	CK	SA	FM-18C	E-7	3	C O/C	C	A					No Indication
15RBC-61	RBCLC Supply to PASS Liquid Sample Cooler	CK	SA	FM-15B	F-7	3A	C O	C	A	CCF	Q			Augmented
15SOV-215	RBCLC Supply to PASS Liquid Sample Cooler	GL	SO	FM-18C	E-7	3	C C	B	P	PI	Y2			
16-1AOV-101A	Drywell Leak Rate Testing Inner Reference Isolation Valve	GA	AO	FM-49A	D-7	2A	C O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
16-1AOV-101B	Drywell Leak Rate Testing Outer Reference Isolation Valve	GA	AO	FM-49A	E-7	2A	C O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
16-1AOV-102A	Torus Leak Rate Testing Outer Reference Isolation Valve	GA	AO	FM-49A	D-7	2A	C O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
16-1AOV-102B	Torus Leak Rate Testing Inner Reference Isolation Valve	GA	AO	FM-49A	C-7	2A	C O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
20AOV-83	Drywell Floor Drain Pumps A&B Discharge Outboard Isolation Valve	BAL	AO	FM-17A	F-6	2A	C O/C	A	A	FC LTJ PI STC	Q RR Y2 Q			
20AOV-95	Drywell Equipment Drain Pump Discharge Inboard Isolation Valve	BAL	AO	FM-17A	C-6	2A	C O/C	A	A	FC LTJ PI STC	Q RR Y2 Q			
20MOV-82	Drywell Floor Drain Sump Pump Discharge Inboard Isolation Valve	GA	MO	FM-17A	F-7	2A	C O	A	A	LTJ PI STC	AJ Y2 RR		 ROJ-34	
20MOV-94	RDW Drywell Equipment Drain Sump Pump Inboard Isol Valve	GA	MO	FM-17A	C-6	2A	C O	A	A	LTJ PI STC	AJ Y2 RR		 ROJ-34	
20RD-18	Penetration X-18 Overpressure Protection Rupture Disc	RPD	SA	FM-17A	H-7	2A	C C	D	A	DT	Y5			
23AOV-42	SPCI Turbine Steam Supply Upstream Drain Isolation Valve	GA	AO	FM-25A	G-2	2	C O	B	A	FC PI STC	Q Y2 Q			
23EFV-01A	HPCI Turbine Steam Supply Leak Detection X-51C Excess Flow Check Valve	CK	SA	FM-25A	G-6	1	C O	A/C	A	SC LT	RR SO	VRR-02 VRR-02		

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
23EFV-01B	HPCI Turbine Steam Supply Leak Detection X-51D Excess Flow Check Valve	CK	SA	FM-25A	G-7	1	C O	A/C	A	SC LT	RR SO	VRR-02 VRR-02		
23EFV-02A	HPCI Turbine Steam Supply Leak Detection X-27C Excess Flow Check Valve	CK	SA	FM-25A	G-7	1	C O	A/C	A	SC LT	RR SO	VRR-02 VRR-02		
23EFV-02B	HPCI Turbine Steam Supply Leak Detection X-27D Excess Flow Check Valve	CK	SA	FM-25A	G-7	1	C O	A/C	A	SC LT	RR SO	VRR-02 VRR-02		
23HPI-12	HPCI Turbine Exhaust Check Valve	CK	SA	FM-25A	C-6	2	O/C C	A/C	A	DIS CCF COF LTJ	RR RR Q AJ		ROJ-12	CVCM JAF-08
23HPI-13	HPCI Drain Pot Drain to Torus Stop Check Valve	CK	SA, MA	FM-25A	C-7	2	O/C C	C	A	LTJ CCF COF	AJ RR Q		ROJ-26 ROJ-13	
23HPI-130	HPCI Gland Seal Cooling Return Check Valve	CK	SA	FM-25A	C-5	2	O/C C	C	A	DIS	RR		ROJ-17	
23HPI-131	HPCI Condensate Pump P-141 Discharge Check Valve	CK	SA	FM-25A	C-5	2	C C	C	A	DIS	RR		ROJ-18	CVCM JAF-20A
23HPI-18	HPCI Pump Discharge to Reactor Check Valve	CK	SA	FM-25A	F-7	1	O C	C	A	SC	CS		CSJ-03	CVCM JAF-20B
23HPI-32	HPCI Booster Pump P-1B Suction From CST 33TK-12A and B Check Valve	CK	SA	FM-25A	G-5	2	C C	C	A	DIS	RR		ROJ-14	CVCM JAF-21A
23HPI-402	HPCI Exhaust Vacuum Breaker Check Valve	CK	SA	FM-25A	E-7	2A	O/C C	C	A	CCF COF	RR RR		ROJ-23 ROJ-23	
23HPI-403	HPCI Exhaust Vacuum Breaker Check Valve	CK	SA	FM-25A	E-7	2A	O/C C	C	A	CCF COF	RR RR		ROJ-23 ROJ-23	
23HPI-56	HPCI Pot Drain to Torus Check Valve	CK	SA	FM-25A	C-6	2	O C	C	A	DIS	RR		ROJ-13	
23HPI-61	HPCI Booster Pump P-1B suction from Suppression Pool Check Valve	CK	SA	FM-25A	B-7	2	O C	C	A	DIS	RR		ROJ-15	CVCM JAF-21B
23HPI-62	HPCI Mi Flow Line to HRH Check Valve	CK	SA	FM-25A	F-4	2	O C	C	A	DIS	RR		ROJ-16	CVCM JAF-22
23HPI-65	HPCI Turbine Exhaust To Torus Check Valve	CK	SA	FM-25A	C-6	2	O/C C	A/C	A	LTJ CCF COF	AJ RR Q		ROJ-12	CVCM JAF-08
23MOV-14	HPCI Turbine Steam Supply Isolation Valve	GA	MO	FM-25A	F-3	2	O C	B	A	PI STO	Y2 Q			
23MOV-15	HPCI Steam Supply Outboard Isolation Valve	GA	MO	FM-25A	F-8	1	O/C C	A	A	LTJ PI STC STO	AJ Y2 RR RR		ROJ-31 ROJ-31	

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
23MOV-16	HPCI Steam Supply Outboard Isolation Valve	GA	MO	FM-25A	F-7	1	O/C O	A	A	LTJ PI STC STO	AJ Y2 Q Q			
23MOV-17	HPCI Booster Pump P-1B Suction From CST 33TK-12 and Isolation Valve	GA	MO	FM-25A	G-5	2	C O	B	A	PI STC STO	Y2 Q Q			
23MOV-19	HPCI Pump to Reactor Inboard Isolation Valve	GA	MO	FM-25A	F-6	1	O/C C	A	A	LTJ PI STC STO	AJ Y2 Q Q			
23MOV-20	HPCI Pump to Reactor Outboard Isolation Valve	GA	MO	FM-25A	F-6	2	O O	B	A	PI STO	Y2 Q			
23MOV-21	HPCII full Flow Test Return To CST 33TK-12A & B UPSTR Valve	GL	MO	FM-25A	G-6	2	C C	B	A	PI STC	Y2 Q			
23MOV-25	HPCI Main Pump P-1M Min Flow Isolation Valve	GL	MO	FM-25A	F-5	2	O/C C	B	A	PI STC STO	Y2 Q Q			
23MOV-57	HPCI Booster Pump P-1B Suction from Suppression Pool DNSTR ISOL Valve	GA	MO	FM-25A	F-5	2	O C	B	A	PI STO	Y2 Q			
23MOV-58	HPCI Booster Pump P-1B Suction from Suppression Pool UPSTR ISOL Valve	GA	MO	FM-25A	C-7	2	O/C O	B	A	PI STC STO	Y2 Q Q			
23MOV-59	HPCI Turbine Exhaust Line Vacuum Breaker Valve	GA	MO	FM-25A	E-7	2	O O	B	P	PI	Y2			
23MOV-60	HPCI Steam Supply Outboard MOV-16 Bypass Valve	GL	MO	FM-25A	F-7	1	C O	A	A	LTJ PI STC STO	AJ Y2 Q Q			
23SV-34	HPCI Booster Pump P-1B Suction Safety Valve	RV	SA	FM-25A	E-5	2	C C	C	A	RLF	Y10			Y8 Group of two
23SV-66	HPCI Booster Pump P-1B Recirc Safety Valve	RV	SA	FM-25A	D-5	2	C C	C	A	RLF	Y10			Y8 Group of two
23Z-7	HPCI Turbine Exhaust Rupture Disc	RPD	SA	FM-25A	F-3	2	C C	C	A	RT	Y5			
23Z-8	HPCI Turbine Exhaust Rupture Disc	RPD	SA	FM-25A	F-3	2A	C C	C	A	RT	Y5			Augmented
27AOV-101A	Torus Vacuum Breaker VB-6 Isolation Valve	BF	AO	FM-18B	C-6	2A	O/C C	A	A	FC LTJ PI STC STO	Q AJ Y2 Q Q			

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
27AOV-101B	Torus Vacuum Breaker VB-7 Isolation Valve	BF	AO	FM-18B	C-6	2A	O/C C	A	A	FC LTJ PI STC STO	Q AJ Y2 Q Q			
27AOV-111	Drywell Purge and Inert Supply Valve	BF	AO	FM-18B	C-2	2A	C C	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27AOV-112	Drywell Purge and Inert Supply Valve	BF	AO	FM-18B	C-3	2A	C C	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27AOV-113	Drywell Exhaust Inner Isolation Valve	BF	AO	FM-18B	D-8	2A	C C	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27AOV-114	Drywell Exhaust Outer Isolation Valve	BF	AO	FM-18B	D-8	2A	C C	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27AOV-115	Torus Purge and Inert Supply Valve	BF	AO	FM-18B	C-2	2A	C C	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27AOV-116	Torus Purge and inert Isolation Valve	BF	AO	FM-18B	C-3	2A	O/C C	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27AOV-117	Torus Exhaust Inner Isolation Valve	BF	AO	FM-18B	B-8	2A	C C	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27AOV-118	Torus Exhaust Outer Isolation Valve	BF	AO	FM-18B	B-8	2A	C C	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27AOV-126A	Ambient Vaporizer A Inlet Valve	GL	AO	FM-18A	G-5	2A	O O	A	A	FC LTJ PI STC	Q AJ Y2 Q			Augmented
27AOV-126B	Ambient Vaporizer B Inlet Valve	GL	AO	FM-18A	F-5	2A	O C	A	A	FC LTJ PI STC	Q AJ Y2 Q			Augmented

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
27AOV-128A	CAD Train A Nitrogen Make-Up Supply Valve	GL	AO	FM-18A	G-4	2A	O/C C	A	A	FC PI STO	Q Y2 Q			Augmented
27AOV-128B	CAD Train B Nitrogen Make-Up Supply Valve	GL	AO	FM-18A	E-4	2A	O/C C	A	A	FC PI STO	Q Y2 Q			Augmented
27AOV-129A	Drywell PCV and Instrument CAD Train A Backup Valve	GL	AO	FM-18A	F-4	2A	O/C O	A	A	FC PI STC STO	Q Y2 Q Q			Augmented
27AOV-129B	Drywell PCV and Instrument CAD Train B Backup Valve	GL	AO	FM-18A	F-4	2A	O/C C	A	A	FC PI STC STO	Q Y2 Q Q			Augmented
27AOV-131A	CAD Train A Nitrogen Make-Up Isolation Valve	GL	AO	FM-18B	C-4	2	O/C C	A	A	FC PI STC STO	Q Y2 Q Q			
27AOV-131B	CAD Train B Nitrogen Make-Up Isolation Valve	GL	AO	FM-18B	C-3	2	O/C C	A	A	FC LTJ PI STC STO	Q AJ Y2 Q Q			
27AOV-132A	CAD Train A Torus Nitrogen Make-Up Isolation Valve	GL	AO	FM-18B	C-4	2	O/C C	A	A	FC LTJ PI STC STO	Q AJ Y2 Q Q			
27AOV-132B	CAD Train A Torus Nitrogen Make-Up Isolation Valve	GL	AO	FM-18B	C-3	2	O/C C	A	A	FC LTJ PI STC STO	Q AJ Y2 Q Q			
27CAD-19A	Liquid Nitrogen Tank A Outlet Check Valve	CK	SA	FM-18A	G-6	2A	O O/C	C	A	CCF COF	Q, M1 Q			Augmented
27CAD-19B	Liquid Nitrogen Tank B Outlet Check Valve	CK	SA	FM-18A	C-6	2A	O O/C	C	A	CCF COF	Q, M1 Q			Augmented
27CAD-67	CAD Train A Torus Nitrogen Make-Up Check Valve	CK	SA	FM-18B	C-6	2	O/C O	A/C	A	LTJ CCF COF	AJ RR Q		ROJ-32	
27CAD-68	CAD Train A Drywell Nitrogen Make-Up Check Valve	CK	SA	FM-18B	C-4	2	O/C O	A/C	A	LTJ CCF COF	AJ RR Q		ROJ-32	

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
27CAD-69	CAD Train B Drywell Nitrogen Make-Up Check Valve	CK	SA	FM-18B	C-4	2	O/C O	A/C	A	LTJ CCF COF	AJ RR Q		ROJ-32	
27CAD-70	CAD Train B Torus Nitrogen Make-Up Check Valve	CK	SA	FM-18B	C-3	2	O/C O	A/C	A	LTJ CCF COF	AJ RR Q		ROJ-32	
27MOV-113	Drywell Exhaust Isolation Valves 27aOV-113 & 114 Outer Bypass Valve	BTF	MO	FM-18B	C-8	2	O/C C	A	A	LTJ PI STC STO	AJ Y2 Q Q			
27MOV-117	Torus Exhaust Isolation Valves 27AOV-117 & 118 Inner Bypass Valve	BF	MO	FM-18B	B-8	2	O/C C	A	A	LTJ PI STC STO	AJ Y2 Q Q			
27MOV-120	Containment Exhaust to Standby Gas Treatment Isolation Valve	BF	MO	FM-18B	H-8	2	O C	B	A	PI STC STO	Y2 Q Q			
27MOV-121	Containment Exhaust to Standby Gas Treatment Isolation Valve	BF	MO	FM-18B	H-8	2	O C	B	A	PI STC STO	Y2 Q Q			
27MOV-122	Drywell Exhaust Isolation Valves 27AOV-113 & 114 Inboard Bypass Valve	GL	MO	FM-18B	C-8	2	O/C C	A	A	LTJ PI STC STO	AJ Y2 Q Q			
27MOV-123	Drywell Exhaust Isolation Valves 27AOV-113 & 114 Outboard Bypass Valve	GL	MO	FM-18B	B-8	2	O/C C	A	A	LTJ PI STC STO	AJ Y2 Q Q			
27RD-1A	CAD Train A Valve Operating Supply Line Rupture Disc	RPD	SA	FM-18A	F-7	2A	C C	D	A	DT	Y10			Augmented
27RD-1B	CAD Train B Valve Operating Supply Line Rupture Disc	RPD	SA	FM-18A	C-7	2A	C C	D	A	DT	Y10			Augmented
27RD-2A	Pressure Building Coil A Outlet Rupture Disc	RPD	SA	FM-18A	F-6	2A	C C	D	A	DT	Y10			Augmented
27RD-2B	Pressure Building Coil B Outlet Rupture Disc	RPD	SA	FM-18A	C-6	2A	C C	D	A	DT	Y10			Augmented
27SOV-119E1	Containment Analyzer A Torus Sample Outer Isolation Valve	GL	SO	FM-18D	C-7	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-119E2	Containment Analyzer A Torus Sample Inner Isolation Valve	GL	SO	FM-18D	C-6	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
27SOV-119F1	Containment Analyzer B Torus Sample Inner Isolation Valve	GL	SO	FM-18D	D-4	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-119F2	Containment Analyzer B Torus Sample Outer Isolation Valve	GL	SO	FM-18D	C-5	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-120E1	Containment Analyzer A Drywell 310EL Sample Outer Isolation Valve	GL	SO	FM-18D	F-6	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-120E2	Containment Analyzer A Drywell 310EL Sample Inner Isolation Valve	GL	SO	FM-18D	F-6	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-120F1	Containment Analyzer B Drywell 310EL Sample Outer Isolation Valve	GL	SO	FM-18D	F-4	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-120F2	Containment Analyzer B Drywell 310EL Sample Inner Isolation Valve	GL	SO	FM-18D	F-4	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-122E1	Containment Analyzer A Drywell 343EL Sample Outer Isolation Valve	GL	SO	FM-18D	F-6	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-122E2	Containment Analyzer A Drywell 343EL Sample Inner Isolation Valve	GL	SO	FM-18D	F-6	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-122F1	Containment Analyzer B Drywell 343EL Sample Outer Isolation Valve	GL	SO	FM-18D	G-4	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-122F2	Containment Analyzer B Drywell 343EL Sample Inner Isolation Valve	GL	SO	FM-18D	G-4	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-123E1	Containment Analyzer A Drywell Sample Outer Isolation Valve	GL	SO	FM-18D	E-6	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
27SOV-123E2	Containment Analyzer A Drywell Sample Inner Isolation Valve	GL	SO	FM-18D	E-6	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-123F1	Containment Analyzer B Drywell 276EL Sample Outer Isolation Valve	GL	SO	FM-18D	F-4	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-123F2	Containment Analyzer B Drywell 276EL Sample Inner Isolation Valve	GL	SO	FM-18D	F-4	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-124E1	Containment Analyzer A Post Accident Sampling Return Header Outer Isolation Valve	GL	SO	FM-18D	C-4	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-124E2	Containment Analyzer A Post Accident Sampling Return Header Inner Isolation Valve	GL	SO	FM-18D	C-4	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-124F1	Containment Analyzer B Post Accident Sample Return Outer Isolation Valve	GL	SO	FM-18D	C-4	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-124F2	Containment Analyzer B Post Accident Sample Return Inner Isolation Valve	GL	SO	FM-18D	C-4	2	C C/O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-125A	Drywell Rad Monitor 17-04-1 Sample Return Inner Isolation Valve	GL	SO	FM-18B	F-5	2	C O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-125B	Drywell Rad Monitor 17-04-2 Sample Return Inner Isolation Valve	GL	SO	FM-18B	F-4	2	C O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-125C	Drywell Rad Monitor 17-04-1 Sample Return Outer Isolation Valve	GL	SO	FM-18B	F-5	2	C O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-125D	Drywell Rad Monitor 17-04-02 Sample Return Outer Isolation Valve	GL	SO	FM-18B	F-4	2	C O	A	A	FC LTJ PI STC	Q AJ Y2 Q			

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
27SOV-135A	Drywell Rad Monitor 17-04-1 Sample Second Isolation Valve	GL	SO	FM-18B	E-5	2	C O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-135B	Drywell Rad Monitor 17-04-2 Sample Outer Isolation Valve	GL	SO	FM-18B	F-5	2	C O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-135C	Drywell Rad Monitor 17-04-1 Sample First Isolation Valve	GL	SO	FM-18B	E-5	2	C O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-135D	Drywell Rad Monitor 17-04-2 Sample Inner Isolation Valve	GL	SO	FM-18B	F-5	2	C O	A	A	FC LTJ PI STC	Q AJ Y2 Q			
27SOV-141	Drywell PCV and Instrument Air Or Normal N2 Cross-Tie Valve	GL	SO	FM-39C	E-6	2	O/C O	A	A	FC LTJ PI STC STO	Q AJ Y2 Q Q			
27SOV-145	DW instrument Nitrogen Backup Supply Isolation Valve	GL	SO	FM-39C	G-5	2	O/C O	A	A	FC LTJ PI STC STO	Q AJ Y2 Q Q			
27SV-114A	CAD Train A Valve Operating Supply Safety Valve	RV	SA	FM-18A	G-6	2A	C C	C	A	RT	Y10			Augmented
27SV-114B	CAD Train B Valve Operating Supply Safety Valve	RV	SA	FM-18A	D-6	2A	C C	C	A	RT	Y10			Augmented
27SV-115A	Ambient A Vaporizer Outlet Safety Valve	RV	SA	FM-18A	G-4	2A	C C	C	A	RT	Y10			Augmented
27SV-115B	Ambient B Vaporizer Outlet Safety Valve	RV	SA	FM-18A	E-4	2A	C C	C	A	RT	Y10			Augmented
27SV-118A	Liquid Nitrogen Tank A Outlet Safety Valve	RV	SA	FM-18A	G-6	2A	C C	C	A	RT	Y10			Augmented
27SV-118B	Liquid Nitrogen Tank B Outlet Safety Valve	RV	SA	FM-18A	C-6	2A	C C	C	A	RT	Y10			Augmented
27SV-119A	Pressure Building Coil A Inlet Safety Valve	RV	SA	FM-18A	F-7	2A	C C	C	A	RT	Y10			Augmented
27SV-119B	Pressure Building Coil B Inlet Safety Valve	RV	SA	FM-18A	C-7	2A	C C	C	A	RT	Y10			Augmented
27SV-201A	Drywell Instrument NW Normal Supply Safety Valve	RV	SA	FM-18A	F-3	2A	C C	C	A	RT	Y10			Augmented

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
27SV-201B	Drywell Instrument N2 Normal Supply Safety Valve	RV	SA	FM-18A	F-3	2A	C C	C	A	RT	Y10			Augmented
27SV-202	Drywell Instrument N2 Backup Supply Safety Valve	RV	SA	FM-18A	H-3	2A	C C	C	A	RT	Y10			Augmented
27VB-1	Torus Downcomer Vacuum Breaker	RV	SA	FM-18B	C-6	2	O/C C	A/C	A	LTJ LTL PI SORT	AJ RR Y2 Y2			
27VB-2	Torus Downcomer Vacuum Breaker	RV	SA	FM-18B	C-6	2	O/C C	A/C	A	LTJ LTL PI SORT	AJ RR Y2 Y2			
27VB-3	Torus Downcomer Vacuum Breaker	RV	SA	FM-18B	C-6	2	O/C C	A/C	A	LTJ LTL PI SORT	AJ RR Y2 Y2			
27VB-4	Torus Downcomer Vacuum Breaker	RV	SA	FM-18B	C-6	2	O/C C	A/C	A	LTJ LTL PI SORT	AJ RR Y2 Y2			
27VB-5	Torus Downcomer Vacuum Breaker	RV	SA	FM-18B	C-6	2	O/C C	A/C	A	LTJ LTL PI SORT	AJ RR Y2 Y2			
27VB-6	Reactor Building To Torus Vacuum Breaker	RV	SA	FM-18B	C-6	2	O/C C	A/C	A	LTJ PI SORT	AJ Y2 Y2			
27VB-7	Reactor Building To Torus Vacuum Breaker	RV	SA	FM-18B	C-6	2	O/C C	A/C	A	LTJ PI SORT	AJ Y2 Y2			
29AOV-80A	Main Steam Line A Inboard Isolation	GL	AO	FM-29A	E-5	1	C O	A	A	FC LTJ PI STC	RO AJ Y2 CS		ROJ-19 CSJ-08	
29AOV-80B	Main Steam Line B Inboard Isolation	GL	AO	FM-29A	D-5	1	C O	A	A	FC LTJ PI STC	RO AJ Y2 CS		ROJ-19 CSJ-08	
29AOV-80C	Main Steam Line C Inboard Isolation	GL	AO	FM-29A	D-5	1	C O	A	A	FC LTJ PI STC	RO AJ Y2 CS		ROJ-19 CSJ-08	

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
29AOV-80	Main Steam Line D Inboard Isolation	GL	AO	FM-29A	D-5	1	C O	A	A	FC LTJ PI STC	RO AJ Y2 CS		ROJ-19 CSJ-08	
29AOV-86A	Main Steam Line A Outboard Isolation	GL	AO	FM-29A	G-4	1	C O	A	A	FC LTJ PI STC	RO AJ Y2 CS		CSJ-05 CSJ-08	
29AOV-86B	Main Steam Line B Outboard Isolation	GL	AO	FM-29A	F-4	1	C O	A	A	FC LTJ PI STC	RO AJ Y2 CS		CSJ-05 CSJ-08	
29AOV-86C	Main Steam Line C Outboard Isolation	GL	AO	FM-29A	E-4	1	C O	A	A	FC LTJ PI STC	RO AJ Y2 CS		CSJ-05 CSJ-08	
29AOV-86D	Main Steam Line D Outboard Isolation	GL	AO	FM-29A	D-4	1	C O	A	A	FC LTJ PI STC	RO AJ Y2 CS		CSJ-05 CSJ-08	
29EFV-30A	Main Steam Line B Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-30B	Main Steam Line B Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-30C	Main Steam Line A Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-30D	Main Steam Line A Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-34A	Main Steam Line D Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-34B	Main Steam Line D Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-34C	Main Steam Line C Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-34D	Main Steam Line C Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-53A	Main Steam Line C Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-53B	Main Steam Line C Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-53C	Main Steam Line D Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-53D	Main Steam Line D Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
29EFV-54A	Main Steam Line A Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-54B	Main Steam Line A Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-54C	Main Steam Line B Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29EFV-54D	Main Steam Line B Excess Flow	CK	SA	FM-29A	E-8	1	C O	A	A/C	SC LT	RO RO	VRR-02 VRR-02		
29MOV-200A	Main Steam Leakage Collection Line A Isolation	GL	MO	FM-29A	C-3	2A	O C	B	A	PI STO	Y2 Q			Augmented
29MOV-200B	Main Steam Leakage Collection Line B Isolation	GL	MO	FM-29A	B-3	2A	O C	B	A	PI STO	Y2 Q			Augmented
29MOV-201A	Main Steam Leakage Collection Line A Upstream Isolation	GL	MO	FM-29A	C-3	2A	O/C C	B	A	PI STC STO	Y2 Q Q			Augmented
29MOV-201B	Main Steam Leakage Collection Line B Upstream Isolation	GL	MO	FM-29A	B-3	2A	O/C C	B	A	PI STC STO	Y2 Q Q			Augmented
29MOV-202A	Main Steam Leakage Collection Line A Downstream Isolation	GL	MO	FM-29A	C-3	2A	O/C C	B	A	PI STC STO	Y2 Q Q			Augmented
29MOV-202B	Main Steam Leakage Collection Line B Downstream Isolation	GL	MO	FM-29A	B-3	2A	O/C C	B	A	PI STC STO	Y2 Q Q			Augmented
29MOV-203A	Main Steam MSIV Stem Leakoff Line B Isolation	GL	MO	FM-29A	C-3	2A	O C	B	A	PI STO	Y2 Q		CSJ-06	Augmented
29MOV-203B	Main Steam MSIV Stem Leakoff Line B Isolation	GL	MO	FM-29A	H-3	2A	O C	B	A	PI STO	Y2 CS		CSJ-06	Augmented
29MOV-204A	Main Steam Leakage Collection Line A Bypass Isolation	GL	MO	FM-29A	C-3	2A	C O	B	A	PI STO	Y2 CS			Augmented
29MOV-204B	Main Steam Leakage Collection Line B Bypass Isolation	GL	MO	FM-29A	B-3	2A	C O	B	A	PI STC	Y2 Q			Augmented
29MOV-74	Main Steam Line Drain Inboard Isolation	GL	MO	FM-29A	C-6	1	C C	A	A	LTJ PI STC	AJ Y2 RR		ROJ-35	
29MOV-77	Main Steam Line Drain Outboard Isolation	GL	MO	FM-29A	C-5	1	C C	A	A	LTJ PI STC	AJ Y2 RR			
34FWS-28A	Feedwater Supply Line A Inboard Check	CK	SA	FM-34A	E-7	1	C O	A/C	A	LTJ CCF CCO	Q RR RR		ROJ-20 ROJ-20	
34FWS-28B	Feedwater Supply Line B Inboard Check	CK	SA	FM-34A	E-7	1	C O	A/C	A	LTJ CCF CCO	Q RR RR		ROJ-20 ROJ-20	

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
34NRV-111A	Feedwater Supply Line A Outboard Check	CK	AO, SA	FM-34A	E-7	1	C O	A/C	A	LTJ PI CCF COF	Q Y2 RR RR		 ROJ-30 ROJ-30	
34NRV-111B	Feedwater Supply Line B Outboard Check	CK	AO, SA	FM-34A	E-7	1	C O	A/C	A	LTJ PI CCF CCO	Q Y2 RR RR		 ROJ-30 ROJ-30	
39IAS-22	DW Instrument Air/N2 Supply Check Valve	CK	SA	FM-39C	E-5	2	O/C O/C	A/C	A	COF LTJ	RO AJ		ROJ-21	
39IAS-29	Drywell Instrument N2 Supply Check Valve	CK	SA	FM-39C	F-3	2	O/C O/C	A/C	A	COF LTJ	RO AJ		ROJ-21	
46(70)ESW-101	CR/RR Vent 70AHU-3A & 12A ESW Supply Isolation Valve	GA	MA	FB-35E	G-6	3	O C	B	A	SO	Y2			
46(70)ESW-102	CR/RR Vent 70AHU-3B & 12B ESW Supply Isolation Valve	GA	MA	FB-35E	C-6	3	O C	B	A	SO	Y2			
46(70)ESW-103	CR/RR Vent 70AHU-3A & 12A ESW Return Isolation Valve	GA	MA	FB-35E	F-6	3	O C	B	A	SO	Y2			
46(70)ESW-104	CR/RR Vent 70AHU-3B & 12B ESW Return Isolation Valve	GA	MA	FB-35E	C-6	3	O C	B	A	SO	Y2			
46(70)SWS-101	CR/RR Vent SW to System A Service Water Supply Check Valve	CK	SA	FB-35E	H-8	3	C O	C	A	CCF COF	Q Q			 CVCM JAF-02
46(70)SWS-102	CR/RR Vent SW to System B Service Water Supply Check Valve	CK	SA	FB-35E	H-8	3	C O	C	A	CCF COF	Q Q			 CVCM JAF-02
46(70)SWS-13	CR/RR Chiller 2a Condenser Service Water Supply Isol Valve	GL	MA	FB-35E	H-4	3	C LO	B	A	SC	Y2			
46(70)SWS-14	CR/RR Chiller 2B Condenser Service Water Supply Isol Valve	GL	MA	FB-35E	E-4	3	C LO	B	A	SC	Y2			
46ESW-1A	Emergency Service Water Pump A Discharge Check Valve	CK	SA	FM-46B	E-7	3	O O	C	A	CCF COF	Q Q			
46ESW-1B	Emergency Service Water Pump B Discharge Check Valve	CK	SA	FM-46B	D-7	3	O O	C	A	CCF COF	Q Q			
46ESW-7A	CR/RR ESW Loop A Supply Check Valve	CK	SA	FM-46B	E-5	3	O O	C	A	CCF COF	Q Q			CVCM JAF-13A
46ESW-7B	CR/RR ESW Loop B Supply Check Valve	CK	SA	FM-46B	E-5	3	O O	C	A	CCF COF	Q Q			CVCM JAF-13B
46ESW-9A	ESW Loop A Supply to RB Check Valve	CK	SA	FM-46B	E-4	3	O O	C	A	CCF COF	Q Q			CVCM JAF-14A
46ESW-9B	ESW Loop B Supply to RB Check Valve	CK	SA	FM-46B	D-4	3	O O	C	A	CCF COF	Q Q			CVCM JAF-14B
46MOV-101A	Emergency Service Water Loop A Supply Header Isolation Valve	GA	MO	FM-46B	E-6	3	O C	B	A	PI STO	Y2 Q			

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
46MOV-101B	Emergency Service Water Loop B Supply Header Isolation Valve	GA	MO	FM-46B	C-6	3	O C	B	A	PI STO	Y2 Q			
46MOV-102A	Emergency Service Water Pump A Test Valve	GA	MO	FM-46B	E-6	3	C O	B	A	PI STC	Y2 Q			
46MOV-102B	Emergency Service Water Pump B Test Valve	GA	MO	FM-46B	D-6	3	C O	B	A	PI STC	Y2 Q			
46RD-112A	Emergency Diesel Generator A Jacket Water Cooler ESW Outlet Rupture Disc	RPD	SA	FM-46B	G-7	3	O C	D	A	DT	Y5			
46RD-112B	Emergency Diesel Generator B Jacket Water Cooler ESW Outlet Rupture Disc	RPD	SA	FM-46B	F-6	3	O C	D	A	DT	Y5			
46RD-112C	Emergency Diesel Generator C Jacket Water Cooler ESW Outlet Rupture Disc	RPD	SA	FM-46B	F-7	3	O C	D	A	DT	Y5			
46RD-112D	Emergency Diesel Generator D Jacket Water Cooler ESW Outlet Rupture Disc	RPD	SA	FM-46B	G-6	3	O C	D	A	DT	Y5			
46SWS-60A	East Crescent Unit Coolers SWS Loop A Supply Check Valve	CK	SA	FB-10H	C-5	3	C O	C	A	CCF COF	Q Q			CVCM JAF-15A
46SWS-60B	East Crescent Unit Coolers SWS Loop B Supply Check Valve	CK	SA	FB-10H	C-5	3	C O	C	A	CCF COF	Q Q			CVCM JAF-15B
46SWS-67A	East Electric Bay 67UC-16A Service Water Supply Check Valve	CK	SA	FM-46A	B-6	3	C O	C	A	CCF CCO	Q Q			CVCM JAF-11
46SWS-67B	East Electric Bay 67UC-16B Service Water Supply Check Valve	CK	SA	FM-46A	B-7	3	C O	C	A	CCF CCO	Q Q			CVCM JAF-11
46SWS-68	West Cable Tunnel 67E-11 Service Water Supply Check Valve	CK	SA	FM-46A	B-6	3	C O	C	A	CCF CCO	Q Q			CVCM JAF-12
46SWS-69	East Cable Tunnel 67E-14 Service Water Supply Check Valve	CK	SA	FM-46A	B-8	3	C O	C	A	CCF CCO	Q Q			CVCM JAF-12
67PCV-101	Elec Bay and Cable Tunnel Vent Service Water Return Pressure Control Valve	GL	AO	FM-46A	D-2	3	O O	B	A	FO SO	Q Q			
70TCV-120A	Relay Room Vent AHU-12A Chilled Water Outlet Temp Control Valve	3W	AO	FB-35E	F-7	3	O O	B	A	FO SO	Q Q			
70TCV-120B	Relay Room Vent AHU-12B Chilled Water Outlet Temp Control Valve	3W	AO	FB-35E	C-6	3	O O	B	A	FO SO	Q Q			

Valve ID	Description	Valve Type	Actu Type	P&ID	Coord	Class	Positions Safety Normal	Cat	Active Passive	Test	Freq	Relief Request	Justification	Comments
70TCV-121A	Control Room Vent AHU-3A Chilled Water Outlet Temperature Control Valve	3W	AO	FB-35E	F-6	3	O O	B	A	FO SO	Q Q			
70TCV-121B	Control Room Vent AHU-3A Chilled Water Outlet Temperature Control Valve	3W	AO	FB-35E	C-7	3	O O	B	A	FO SO	Y2			
70WAC-12A	CR/RR Chilled Water loop A Air Separator Tank 20A Inlet Isolation Valve	GA	MA	FB-35E	F-6	3	C O	B	A	SC	Y2			
70WAC-12B	Control Room/Relay Room Chiller Air Separator Tank Outlet	GA	MA	FB-35E	C-6	3	C O	B	A	SC	Y2			
70WAC-5A	CR/RR Chiller A Evaporator Chilled Water Outlet Isol Valve	GA	MA	FB-35E	F-2	3	C O	B	A	SC	Y2			
70WAC-5B	CR/RR Chiller B Evaporator Chilled Water Outlet Isol Valve	GA	MA	FB-35E	D-2	3	C O	B	A	SC	Y2			

ATTACHMENT 16

CHECK VALVE CONDITION MONITORING PLAN INDEX

CVCM PLAN NUMBER	**REV #**	**TITLE**
JAF-02 CMP	Draft	10RHR-64A/B/C/D
JAF-03 CMP	Draft	10RHR-14A/B/C/D
JAF-04 CMP	Draft	11SLC-43A/B
JAF-05 CMP	Draft	13RCIC-4/5
JAF-06 CMP	Draft	14AOV-13A/B
JAF-07 CMP	Draft	14CSP-10A/B
JAF-08 CMP	Draft	23HPI-12/65
JAF-10 CMP	Draft	46(70)SWS-101/102
JAF-11 CMP	Draft	46SWS-67A/B
JAF-12 CMP	Draft	46SWS-68/69
JAF-13A CMP	Draft	46ESW-7A
JAF-13B CMP	Draft	46ESW-7B
JAF-14 CMP	Draft	46ESW-9A/B
JAF-15A CMP	Draft	46SWS-60A
JAF-15B CMP	Draft	46SWS-60B
JAF-20A CMP	Draft	23HPI-130
JAF-20B CMP	Draft	23HPI-131
JAF-21A CMP	Draft	23HPI-32
JAF-21B CMP	Draft	23HPI-61
JAF-22 CMP	Draft	23HPI-62