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

April 14, 2009

Mr. David A. Christian
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNIT NO. 1, FOURTH 10-YEAR INSERVICE INSPECTION PLAN – SYSTEM PRESSURE TESTING (TAC NO. MD9956)

Dear Mr. Christian:

By letter dated October 7, 2008, Virginia Electric and Power Company (the licensee) requested relief from certain requirements of Section XI of the American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME Code), under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a for the fourth 10-year inservice inspection (ISI) interval for North Anna Power Station, Unit No. 1 (NAPS 1). This letter specifically addresses requests SPT-001 through SPT-006 on system pressure testing for Class 1 components. By letter dated March 17, 2009, the licensee withdrew SPT-006.

For Relief Request (RR) Nos. SPT-001, SPT-002, SPT-003 and SPT-004, the Nuclear Regulatory Commission (NRC) staff finds that the ASME Code requirements would impose a hardship without a compensating increase in quality and safety. The licensee's proposed alternative provides reasonable assurance of leak-tight integrity and structural integrity of the subject components. Therefore, the NRC staff authorizes the proposed alternatives pursuant to 10 CFR 50.55a(a)(3)(ii) for the fourth 10-year ISI interval for NAPS 1.

The NRC staff finds that ASME Code requirements are impractical for RR No. SPT-005. The licensee's proposed alternative provides reasonable assurance of structural integrity. Therefore, the requested relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the fourth 10-year ISI interval for NAPS 1. The relief granted is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

All other requirements of the ASME Code, Section XI for which relief has not been specifically requested and approved remain applicable, including a third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions concerning this matter, please contact Donna Wright at (301) 415-1864.

Sincerely,

Melanie C Wong

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-338

Enclosure: Safety Evaluation

cc w/encl: Distribution via Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL PROGRAM

RELIEF REQUEST NOS. SPT-001 THROUGH SPT-005

NORTH ANNA POWER STATION, UNIT NO. 1

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

1.0 INTRODUCTION

By letter dated October 7, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082880160), Virginia Electric and Power Company (the licensee) submitted Relief Requests (RRs) Nos. SPT-001 through SPT-006 on system pressure testing for Class 1 components applicable to North Anna Power Station, Unit No. 1 (NAPS 1) for the fourth 10-year inservice inspection (ISI) interval. By letter dated March 17, 2009 (ADAMS Accession No. ML090771305), the licensee withdrew SPT-006. SPT-003 and SPT-004 pertain to the system boundary that is subject to test pressurization and the American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME Code)-required test pressure during performance of a system leakage test conducted at or near the end of the inspection interval. In lieu of the ASME Code requirement to conduct the test on all ASME Class 1 pressure retaining components within the system boundary, the licensee has proposed an alternative to pressurize up to the inboard isolation valve which would exclude a small segment of the Class 1 pressure boundary from attaining test pressure. The components will be visually examined during a Class 2 system leakage test conducted at a lower pressure for segments of piping. In RRs SPT-001, SPT-002, and SPT-005, the licensee proposes to visually examine for evidence of leakage from the past operating cycle since the components are either not pressurized to ASME Code-required test pressure or the environment is too hostile to perform a visual examination at test pressure.

The Nuclear Regulatory Commission (NRC) staff has evaluated the licensee's RR Nos. SPT-001 through SPT-004 pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a(a)(3)(ii) that compliance to the requirement of the ASME Code would result in hardship without a compensating increase in the level of quality and safety; and RR No. SPT-005 pursuant to 10 CFR 55.55a(g)(6)(i) based on impracticality of performing the ASME Code-required test.

2.0 REGULATORY REQUIREMENTS

Section 50.55a(g) of 10 CFR requires that inservice inspection (ISI) of ASME Code Class 1, 2, and 3 components be performed in accordance with Section XI of the ASME Code and applicable addenda, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i), which, states, that if code requirements are determined impractical by the Commission, relief may be granted and alternatives may be imposed that are authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed. In addition, according to 10 CFR 50.55a(a)(3), alternatives to the requirements of paragraph 50.55a(g) may be used, when authorized by the NRC, if an applicant demonstrates that the proposed alternatives would provide an acceptable level of quality and safety or if the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that ISI of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval subject to the limitations and modifications listed in paragraph (b) of this section. The ISI Code of Record for the fourth 10-year inservice inspection interval of NAPS 1 is the 2004 Edition of the ASME Code, Section XI.

3.0 TECHNICAL EVALUATION

3.1 Relief Request No. SPT-001

(a) System/Component(s) for Which Relief is Requested

Reactor Vessel Bottom Mounted Instrument Tubing Partial Penetration Welds

(b) ASME Code Requirements

Table IWB-2500-1, Examination Category B-P, Item Number B15.10 requires that a system leakage test be performed each refueling outage in accordance with the requirements of IWB - 5220.

(c) Licensee's Request for Relief

Relief is requested from performing the ASME Code required visual (VT-2) examination of the reactor vessel bottom head during system leakage and hydrostatic tests.

(d) Licensee's Basis for Requesting Relief

The licensee stated the following:

> In order to meet the ASME Code, Section XI pressure and temperature requirements for the system leakage test of the reactor vessel, the reactor containment at NAPS 1 is required to be at a sub-atmospheric pressure. Station administrative procedures require that self-contained breathing apparatus be worn for containment entries under these conditions. This requirement significantly complicates the visual (VT-2) visual examination of the bottom of the reactor vessel during testing. Access to the bottom of the reactor vessel requires that the examiner descend several levels by ladder and navigate a small entrance leading

to the reactor vessel. In addition to these physical constraints, the examiner must contend with extreme environmental conditions: elevated air temperature due to reactor coolant temperature above 500 degrees F and limited air circulation in the vessel cubicle. In addition, the examiner is limited to the approximate 30 minute capacity of the breathing apparatus for containment entry, the VT-2 examination, and containment exit.

(e) Licensee's Proposed Alternative

The licensee stated the following:

> The Improved Technical Specifications (ITS) establish limits on [reactor coolant system] RCS leakage to one gallon per minute of unidentified leakage and no identified leakage in the pressure boundary. To monitor for leakage, the ITS[s] also require that (a) one containment sump (level or discharge flow) monitor, and (b) one containment atmosphere radioactivity monitor (gaseous or particulate) be operable during modes 1, 2, 3, and 4. In addition, the plant must verify RCS operational leakage is within limits by performance of an RCS inventory balance at a frequency not exceeding 72 hours. The ITS[s] also require that a channel check be performed of the required containment atmosphere radioactivity monitor at a frequency not exceeding 12 hours. The in-core sump room has a level alarm in the control room requiring operator action. These actions would identify any integrity concerns associated with this area. A VT-2 visual examination will be conducted when the containment is at atmospheric conditions [during] each refueling outage for evidence of boric acid corrosion.

(f) NRC Staff's Evaluation

The NRC staff has reviewed the information concerning the ISI program RR SPT-001, for the fourth 10-year ISI interval of NAPS 1 pertaining to visual VT-2 examination of the bottom of the reactor vessel, including examination of the instrumentation nozzle partial penetration welds. The Code of Record requires that a VT-2 visual examination be conducted during each system leakage test of the reactor coolant system. Since the containment building is at sub-atmospheric condition during the system leakage test, the examiner must wear a self-contained breathing apparatus that limits his work duration and mobility. In addition to these physical constraints, the examiner must contend with high ambient temperatures. Thus, compliance with the ASME Code examination requirement would result in hardship or unusual difficulty to the licensee.

The licensee proposed, as an alternative, to perform a VT-2 visual examination for evidence of boric acid corrosion when the containment is at atmospheric condition during refueling. In addition, the licensee noted that the ITSs require monitoring of reactor coolant leak rate, atmospheric particulate radioactivity, and containment sump level. The NRC staff believes that the boric acid corrosion inspection performed at the end of the fuel cycle is in itself a reliable inspection for reactor coolant leakage and the VT-2 visual examination for evidence of boric acid corrosion conducted during each refueling outage would, therefore, provide a reasonable assurance of leak-tight integrity. The NRC staff also believes that the licensee's proposed alternative will provide reasonable assurance that unallowable inservice leaks, if developed at the bottom of the reactor vessel, will be detected for appropriate corrective action prior to return of the vessel back to service. The NRC staff has, therefore, determined that the ASME Code-required examinations at the bottom of the reactor vessel during system leakage testing would result in hardship without a compensating increase in the level of quality and safety.

(g) Conclusion

The NRC staff concludes that performance of the ASME Code-required VT-2 visual examination of the bottom of the reactor vessel during the system leakage testing when the containment is at sub-atmospheric condition would result in hardship to the licensee without a compensating increase in the level of quality and safety. The licensee's proposed alternative examination would provide a reasonable assurance of structural integrity of the reactor vessel bottom head including the instrumentation nozzle partial penetration welds. Therefore, the proposed alternative in RR No. SPT-001, is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the fourth 10-year ISI interval of NAPS 1. All other requirements of ASME Code, Section XI for which relief has not been specifically requested remain applicable, including a third-party review by the Authorized Nuclear Inservice Inspector.

3.2 Relief Request No. SPT-002

(a) System/Component(s) for Which Relief is Requested

Reactor Coolant System (RCS) Vent, Drain, Test, Sample (VDTS), and Instrument Connections 1-inch or less in diameter.

(b) ASME Code Requirements

Table IWB-2500-1, Examination Category B-P, Item Numbers B15.10 requires that a system leakage test be performed each refueling outage in accordance with the requirements of IWB-5220. IWB-5222(b) requires that the system leakage test conducted at or near the end of the inspection interval extend to all Class 1 pressure retaining components within the system boundary.

(c) Licensee's Request for Relief

The RCS VDTS and instrument connections will be visually examined for leakage including any evidence of past leakage during each refueling outage and during the end of interval system leakage test with the isolation valves in the normally closed position which corresponds to RCS at nominal operating pressure and at near operating temperature.

(d) Licensee's Basis for Requesting Relief

The licensee stated the following:

> The subject piping segments are equipped with valves that provide for double isolation of the [RCS] pressure boundary. These valves are maintained closed during normal operation and the piping outboard of the first isolation valve is, therefore, not normally pressurized. The proposed alternative provides an acceptable level of quality and safety based on the following:

1. ASME Section XI Code, paragraph IWA-4540, provides the requirements for pressure testing of piping and components after repairs by welding to the pressure boundary. IWA-4540(b)(6) excludes component connections, piping, and associated valves that are 1-inch nominal pipe size and smaller from the pressure test. Consequently, pressure testing and the associated visual examination of [the1-inch or less in diameter] RCS [VDTS] connections once each 10-year interval is unwarranted considering that a repair/replacement weld on the same connections is exempted by the ASME Section XI Code.

2. The non-isolable portion of the RCS vent and drain connections will be pressurized and visually examined as required. Only the normally isolable portion of these small diameter vent and drain connections will not be pressurized.

   The Improved Technical Specifications (ITS) require RCS leakage monitoring (TS 4.4.6.2.1) during normal operation. Should any of the ITS limits be exceeded, then appropriate corrective actions, which may include shutting the plant down, are required to identify the source of leakage and restore the RCS boundary integrity.

   During [the] 1998, [NAPS 1] refueling outage similar piping segments were pressurized by removing a flange and connecting a test rig. A majority of these piping segments are located in close proximity to the RCS main loop piping thus requiring personnel entry into high radiation areas within the containment. The dose associated with this testing was 1.5 man-rem.

(e) <u>Licensee's Proposed Alternative</u>

The licensee stated the following:

> As an alternative to the [ASME] Code-required pressure test of the subject Class 1 [RCS] pressure boundary connections, the following is proposed:
>
> 1. The RCS [VDTS and instrument] connections will be visually examined for leakage and any evidence of past leakage, with the isolation valves in the normally closed position each refueling outage during the [ASME Code] Class 1 system leakage test (IWB-5221 and IWB-5222(a)).
>
> 2. The RCS [VDTS and instrument] connections will also be visually examined with the isolation valves in the normally closed position during the 10-year ISI pressure test (IWB - 5221 and IWB-5222(a)). This examination will be performed with the RCS at nominal operating pressure and at near operating temperature.

(f) <u>NRC Staff's Evaluation</u>

For NAPS 1, the ASME Code, Section XI requires that all Class 1 components within the RCS undergo a system leakage test at the end of each refueling outage and a system hydrostatic test at or near the end of each inspection interval. In RR No. SPT-002, the licensee proposed an alternative to the ASME Code requirement of the test for the RCS VDTS and instrument connections which would isolate a segment of piping between the inboard and outboard isolation valves from being pressurized during a system leakage test. The pipe segments include two manually operated valves separated by a short pipe that is connected to the reactor coolant system. The line configuration, as outlined, provides double-isolation of the RCS. Under normal plant operating conditions, the subject pipe segments would experience RCS temperature and pressure only if leakage through the inboard isolation valves occurs. For the licensee to perform the ASME Code-required test, it would be necessary to manually open the inboard valves to pressurize the pipe segments. Pressurization by this method would preclude the RCS double valve isolation and may cause safety concerns for the personnel performing the examination. Typical line/valve configurations are in close proximity of the RCS main runs of pipe and thus, would require personnel entry into high radiation areas within the containment. Manual actuation (opening and closing) of these valves is estimated to expose plant personnel to approximately 1.5 man-rem per test. This estimate was based on the 1998, NAPS 1 refueling outage during which similar piping was tested. The licensee proposed to visually examine the isolation valves in the normally closed position for leaks which would indicate any evidence of past leakage during the operating cycle. Also, the RCS VDTS and instrument connections will be visually examined with the isolation valves in the normally closed position during the 10-year system hydrostatic test.

The NRC staff believes that the licensee's proposed alternative will provide reasonable assurance of structural integrity for the RCS VDTS and instrument connections while maintaining personnel radiation exposure to as low as reasonably achievable (ALARA). The NRC staff has further determined that compliance to the ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

(g) Conclusion

Based on the NRC staff's evaluation of RR No. SPT-002, the licensee's proposed alternative provides reasonable assurance of structural integrity, and compliance with the ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative in RR No. SPT-002 is authorized for the fourth 10-year intervals of NAPS 1. All other requirements of ASME Code, Section XI for which relief has not been specifically requested remain applicable, including a third-party review by the Authorized Nuclear Inservice Inspector.

3.3 Relief Request No. SPT-003

(a) System/Component(s) for Which Relief is Requested

Residual Heat Removal (RHR) System Components and Piping between 1-RH-MOV-1700 and 1-RH-MOV-1701 (RHR suction) on drawing 11715-CBM-094A-3, sheet 1 of 2.

(b) ASME Code Requirements

Table IWB-2500-1, 15.50 (Piping) requires that a system leakage test be conducted at or near the end of each inspection interval. IWB-5221(a) states that the system leakage test be conducted at a pressure not less than the pressure corresponding to 100-percent rated reactor power. In IWB-5222(b), the pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval is required to extend to all Class 1 pressure retaining components within the system boundary.

(c) Licensee's Request for Relief

The licensee proposes an alternative visual examination of the segment of Class 1 piping between 1-RH-MOV-1700 and 1-RH-MOV-1701 (RHR suction) including the valves during the Class 2 system pressure test.

(d) Licensee's Basis for Requesting Relief

The licensee stated the following:

> Normal reactor coolant pressure at 100-percent rated power is approximately 2235 psig. The [subject piping] is separated from this reactor coolant pressure by a single closed valve, and as such does not normally see this pressure. Opening valve 1-RH-MOV-1700 is prevented by a pressure interlock when the pressure in the RCS is above 418 psig. The interlock protects the low pressure RHR system from being over-pressurized by the higher pressure RCS.

There is no provision to pressurize the segment of piping between the two motor-operated valves to the ASME Code-required test pressure using an external source.

(e) Licensee's Proposed Alternative

The segment of piping between the two motor-operated valves including the valves will be visually examined (VT-2) during each refueling outage in operation for evidence of leakage as part of normal Class 1 system leakage test. Additionally, the subject piping and the valves will be visually examined during the Class 2 system pressure test of the adjoining Class 2 piping conducted each inspection period.

(f) NRC Staff's Evaluation

The 2004, ASME Code, Section XI, Table IWB-2500-1, Category B-P, Item Numbers B15.51 and B15.71 require a system leakage test of Class 1 pressure retaining piping and valves once per 10-year interval. The system leakage test is required to be performed at a test pressure not less than the normal operating pressure of the reactor coolant system corresponding to 100-percent rated reactor power and shall include all Class 1 components within the RCS boundary. In RR No. SPT-003, the licensee proposed an alternative to the boundary subject to test pressurization required under the ASME Code for the RHR system between an inboard and an outboard isolation valve in the system boundary. The line configuration, as outlined, provides double-isolation of the RCS. Under normal plant operating conditions, the subject pipe segment would experience RCS temperature and pressure only if leakage through an inboard isolation valve occurs. As requested in SPT-003, with the inboard isolation valve closed during the system leakage test, the segment of piping between an inboard and an outboard isolation valve would not get pressurized to the required test pressure during a system leakage test. To perform the ASME Code-required test, it would be necessary to manually open each inboard isolation valve to pressurize the pipe segment. Pressurization by this method would preclude double valve isolation of the RCS and may cause safety concerns for the personnel performing the examination. Alternatively, the line segment between the isolation valves could be separately pressurized to the required test pressure by an external pump but there is no test connection between the isolation valves to attach a pump.

The subject isolation valves are located inside the containment, and any manual actuation (opening and closing) of these valves would expose plant personnel to undue radiation exposure during modification and restoration of system lineups. The NRC staff finds that compliance with the ASME Code requirement would result in hardship without a compensating increase in the level of quality and safety. The licensee has proposed an alternative to visually examine (VT-2) for leaks in the isolated portion of the subject segments of piping with the inboard and outboard isolation valves in the normally closed position which would indicate any evidence of past leakage during the operating cycle as well as any active leakage during the system leakage test if the inboard isolation valve leaks. One factor that supports the acceptability of the licensee's proposed alternative is that the segment of Class 1 pressure boundary between the inboard and outboard isolation valves in the RHR system that is not tested to the ASME Code-required test pressure, would be pressure-tested at the associated system's operating pressure during the RHR system inservice test during the refueling outage. Another mitigating factor in accepting the test pressure at system operating pressure in lieu of the ASME Code-required test pressure is based on the fact that there is no known degradation mechanism, such as intergranular stress corrosion cracking (IGSCC), primary water stress corrosion cracking (PWSCC), or thermal fatigue, that is likely to affect the welds in the subject segment.

The NRC staff believes that the licensee's proposed alternative will provide reasonable assurance of structural integrity for the piping segments in the RHR system between an inboard and an outboard isolation valve including the valves while maintaining personnel radiation exposure to ALARA.

(g) Conclusion

The NRC staff concludes that test pressurization during system leakage test of the Class 1 pressure retaining components within the system boundary of the RHR system, between an inboard and an outboard isolation valve including the valves, as required by the ASME Code would result in hardship to the licensee without a compensating increase in the level of quality and safety. The licensee's proposed alternative in RR No. SPT-003 provides a reasonable assurance of structural integrity for the subject piping segment. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative in SPT-003 is authorized for the fourth 10-year ISI interval of NAPS 1. All other requirements of the ASME Code, Section XI for which relief has not been specifically requested remain applicable, including a third-party review by the Authorized Nuclear Inservice Inspector

3.4 Relief Request No. SPT-004

(a) System/Component(s) for Which Relief is Requested

Safety Injection (SI) System Components and Piping between 1-SI-195, 1-SI-197, and 1-SI-199, on drawing 11715-CBM-096B-4, sheet 4 of 4, and 1-SI-MOV-1890C and 1-SI-MOV-1890D on drawing 11715-CBM-096A-4, sheet 2 of 3 (Low Head Safety Injection to the Reactor Coolant Cold Legs)

1-SI-211, 1-SI-209, and 1-SI-213, on drawing 11715-CBM-096B-4, sheet 4 of 4, and 1-SI-MOV-1890A and 1-SI-MOV-1890B, on drawing 11715-CBM-096A-4, sheet 2 of 3 (Low Head Safety Injection to the Reactor Coolant Hot Legs)

(b) ASME Code Requirements

Table IWB-2500-1, 15.50 (Piping) requires that a system leakage test be conducted at or near the end of each inspection interval. IWB-5221(a) states that the system leakage test be conducted at a pressure not less than the pressure corresponding to 100-percent rated reactor power. In IWB-5222(b), the pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval is required to extend to all Class 1 pressure retaining components within the system boundary.

(c) Licensee's Request for Relief

The licensee proposes an alternative visual examination of the segment of Class 1 piping and components identified above during the routine Class 2 system pressure test of the connecting system.

(d) Licensee's Basis for Requesting Relief

Normal reactor coolant pressure at 100-percent rated power is approximately 2235 psig. The subject piping is separated from this reactor coolant pressure by check valves, and as such does not normally experience this pressure. External pressurization would be necessary to meet either the ASME Code hydrostatic test requirement or the system leakage test pressure corresponding to 100-percent reactor power. However, there is no provision to pressurize the segment of piping between the check valves to the ASME Code-required test pressure using an external source. Since check valves would be part of the test boundary, a pressure differential would be required between the RCS and the segment of piping to maintain check valve closure.

Maintaining the differential pressure and ensuring no intrusion of test fluid into the RCS to affect reactivity control is considered to be unusually difficult to meet with no compensating increase in the level of quality and safety.

(e) Licensee's Proposed Alternative

The segment of Class 1 piping between the check valves will be visually examined (VT-2) during the Class 1 system leakage test following each refueling outage for evidence of leakage. Additionally, the segment of piping will also be VT-2 visually examined for leakage during the Class 2 system pressure test of the connecting system during each inspection period and also at or near the end of the inspection interval.

(f) NRC Staff's Evaluation

The 2004, ASME Code, Section XI, Table IWB-2500-1, Category B-P, Item numbers B15.51 and B15.71 requires a system leakage test of Class 1 pressure retaining piping and valves once per 10-year interval. The system leakage test is required to be performed at a test pressure not less than the nominal operating pressure of the reactor coolant system corresponding to 100-percent rated reactor power and shall include all Class 1 components within the RCS boundary. In RR

No. SPT-004, the licensee proposed an alternative to the boundary that is subject to test pressurization required under the ASME Code for the SI system between an inboard and an outboard isolation valve in the system boundary. The line configuration, as outlined, provides double-isolation of the RCS. Under normal plant operating conditions, the pipe segment would experience RCS temperature and pressure only if leakage through an inboard isolation valve occurs. As requested in SPT-004, with the inboard isolation valve closed during the system leakage test, the segment of piping between an inboard and an outboard isolation valve would not get pressurized to the required test pressure during a system leakage test. In order to perform the ASME Code-required test, it would be necessary to manually open each inboard isolation valve to pressurize the pipe segment. Pressurization by this method would preclude double valve isolation of the RCS and may cause safety concerns for the personnel performing the examination. Alternatively, the line segment between the isolation valves could be separately pressurized to the required test pressure by an external pump but there is no test connection between the isolation valves to attach a pump.

Some of the isolation valves are located inside the containment and to make provision for connection of an external pressurization source for the line segments would expose plant personnel to undue radiation exposure during modification and restoration of system lineups. The NRC staff finds that compliance with the ASME Code requirement would result in hardship without a compensating increase in the level of quality and safety. The licensee has proposed an alternative to visually examine (VT-2) for leaks in the subject segments of piping during the Class 2 system pressure test of the adjoining piping conducted during each inspection period which would detect any active leakage. One mitigating factor in accepting the test pressure at system operating pressure in lieu of the ASME Code-required test pressure is based on the fact that there is no known degradation mechanism, such as IGSCC, PWSCC, or thermal fatigue that is likely to affect the welds in the subject segment. The NRC staff believes that the licensee's proposed alternative will provide reasonable assurance of structural integrity for the piping segments in the SI system between an inboard and an outboard isolation valve including the valves while maintaining personnel radiation exposure to ALARA.

(g) <u>Conclusion</u>

The NRC staff concludes that test pressurization during system leakage testing of the Class 1 pressure retaining components within the SI system boundary between an inboard and an outboard isolation valve including the valves as required by the ASME Code of Record would result in hardship to the licensee without a compensating increase in the level of quality and safety. The licensee's proposed alternative in RR No. SPT-004 provides a reasonable assurance of structural integrity for the subject piping segment. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative in SPT-004 is authorized for the fourth 10-year ISI interval of NAPS 1. All other requirements of the ASME Code, Section XI for which relief has not been specifically requested remain applicable, including a third-party review by the Authorized Nuclear Inservice Inspector

3.5 <u>Relief Request No. SPT-005</u>

(a) <u>System/Component(s) for Which Relief is Requested</u>

RCS

Components: Class 1 Components and Piping between 1-RC-R-1 (Reactor Inner O-ring), 1-RC-32 and 1-RC-HCV-1544 on drawing 11715-093A-3, sheet 1 of 3.

(b) ASME Code Requirements

Table IWB-2500-1, 15.50 (Piping) requires that a system leakage test be conducted at or near the end of each inspection interval. IWB-5221(a) states that the system leakage test be conducted at a pressure not less than the pressure corresponding to 100-percent rated reactor power. In IWB-5222(b), the pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval is required to extend to all Class 1 pressure retaining components within the system boundary.

(c) Licensee's Request for Relief

Relief is requested from performing the system leakage test at a pressure corresponding to nominal operating pressure during system operation. The licensee proposed an alternative pressure testing requirement in lieu of the system leakage test required under IWB-5221(a) for the reactor vessel head flange leak detection piping.

(d) Licensee's Basis for Requesting Relief

The licensee stated the following:

> Normal reactor coolant pressure at 100-percent rated power is approximately 2235 psig. The components and piping being addressed are associated with the reactor head and flange leak detection system. They are used to support identification of inner O-ring leakage. An increase in temperature above ambient is an indication of inner O-ring seal leakage. High temperature actuates an alarm. On indication of inner O-ring leakage, the isolation valve in the leak-off line can be closed to put the outer O-ring into the pressure retention mode, and the inner O-ring leak detection system would be pressurized to reactor coolant pressure up to the closed isolation valve.
>
> These lines can only be tested externally, since during normal operation they are separated from RCS pressure by the inner O-ring [in a direction opposite that it was designed for]. This could move the inner O-ring from its normal position against the outer channel wall of the reactor vessel flange potentially affecting the O-ring leak tightness and requiring that maintenance be performed.

(e) Licensee's Proposed Alternative

The reactor head and flange leak detection piping will be visually examined (VT-2) each refueling outage for evidence of leakage. Additionally, the licensee will exercise procedural control on operator action leading to identification of leakage by any alarm due to breach of inner O-ring pressure boundary or by RCS inventory balance calculation.

(f) NRC Staff's Evaluation

Section XI, of the ASME Code of Record, requires that all Class 1 components within the reactor coolant system boundary undergo a system leakage test at or near the end of each inspection interval. The licensee requested relief from performing a system leakage test of the reactor vessel head flange seal leak detection piping at the ASME Code-required test pressure corresponding to the normal reactor coolant pressure at 100-percent rated power. The piping is located between the inner and the outer O-ring seals of the vessel flange and is required during plant operation in order to detect failure of the inner flange seal O-ring. The design of this line makes the ASME Code-required system leakage test impractical either with the vessel head in place or removed. The piping cannot be filled completely with water since it can not be vented to remove entrapped air from the line either with the vessel head in place or removed due to its configuration. If a pressure test were to be performed with the head in place, the space between the inner and the outer O-ring seals would be pressurized. The test pressure would exert a net inward force on the inner O-ring that would tend to push it into the recessed cavities that house the retainer with the possibility of damaging the inner O-ring seal. The configuration of this piping also precludes system pressure testing while the vessel head is removed because the odd configuration of the vessel tap coupled with the high test pressure requirement prevents the tap in the flange from being temporarily plugged or connected to other piping. The opening in the flange is smooth walled, making the effectiveness of a temporary seal very limited. Failure of this seal could possibly cause ejection of the device used for plugging or connecting to the vessel.

To perform the system leakage test in accordance with the ASME Code requirements, the reactor vessel head flange seal leak detection piping would have to be redesigned, fabricated, and installed. This would impose a severe burden on the licensee. The licensee has proposed to perform a VT-2 visual examination of the reactor vessel head flange seal leak detection piping during a refueling outage for evidence of leakage from the past operating cycle. The licensee also has procedural control on annunciation of an alarm in the event of any leakage past the inner O-ring and/or RCS mass balance calculations for which the NRC staff believes that the proposed testing would provide reasonable assurance of structural integrity. Therefore, based on the impracticality of complying with the ASME Code requirements, the proposed relief is granted.

(g) Conclusion

Based on the NRC staff's evaluation, a system leakage test of the reactor vessel head flange seal leak detection piping at the ASME Code-required test pressure corresponding to the normal reactor coolant pressure at 100-percent rated power is impractical and would cause a severe burden on the licensee if the requirements were imposed. The licensee's proposed alternative provides reasonable assurance of structural integrity. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), the proposed alternative in RR No. SPT-005 is granted for the fourth 10-year ISI interval of NAPS 1. The relief granted is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. All other requirements of the ASME Code, Section XI for which relief has not been specifically requested remain applicable, including a third-party review by the Authorized Nuclear Inservice Inspector.

4.0 CONCLUSION

The following summarizes the NRC staff's conclusions based on the technical evaluation discussed above.

With respect to RR SPT-001 though SPT-004, the proposed alternatives will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternatives are authorized for the fourth 10-year ISI interval at NAPS 1.

With respect to RR SPT-005, compliance with the specified ASME Code requirements is impractical. The proposed alternative provides reasonable assurance of structural integrity. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), relief is granted for the fourth 10-year ISI interval at NAPS 1. Granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property, or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in the subject requests for relief remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: P. Patnaik, NRR

Date: April 14, 2009

If you have any questions concerning this matter, please contact Donna Wright at (301) 415-1864.

Sincerely,

/RA/

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-338

Enclosure: Safety Evaluation

cc w/encl: Distribution via Listserv

DISTRIBUTION: Public
LPL2-1 R/F
RidsNrrDorlLpl2-1 Resource
RidsNrrLAMO'Brien Resource
RidsNrrPMNorthAnna Resource
RidsAcrsAcnw_MailCTR Resource
RidsRgn2MailCenter Resource (GMcCoy)
RidsOgcRp Resource
RidsNrrDciCsgb Resource
PPatnaik, NRR

ADAMS Accession No.: ML090890171 * by memo dated

OFFICE	NRR/LPL2-1/PM	NRR/LPD2-1/LA	NRR/CSGB/BC	OGC	NRR/LPL2-1/BC
NAME	DWright	MO'Brien	MYoder*		MWong
DATE	4/6/09	4/6/09	3/23/09	4/13/09	4/14/09

OFFICIAL RECORD COPY