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

November 2, 2010

Mr. David A. Heacock
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 (NAPS), UNIT NO. 2, FOURTH 10-YEAR INSERVICE INSPECTION (ISI) INTERVAL, RELIEF REQUESTS SPT-001 THROUGH SPT-006 (TAC NOS. ME3311 THROUGH ME3316)

Dear Mr. Heacock:

By letter to the U.S. Nuclear Regulatory Commission (NRC), dated February 1, 2010, Virginia Electric and Power Company (the licensee) submitted request for relief SPT-001 through SPT-006 from certain requirements of the American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME Code) at NAPS, Unit No. 2. Specifically, in accordance with Title 10 of the *Code* of *Federal Regulations* (10 CFR), Part 50, Section 50.55a(a)(3)(ii), the licensee proposed an alternative to the pressure test requirements of ASME Code, Section XI, IWA-4540(c), for repair and replacement activities for mechanical joints made in the installation of pressure-retaining items. The licensee requested implementation of this alternative during fourth 10-year ISI interval scheduled to start on December 14, 2010, and end on December 13, 2020.

Based on the review of the information the licensee provided, the NRC staff concludes that the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety and the alternatives provide reasonable assurance of structural integrity. Therefore, the licensee's proposed alternative is authorized in accordance with 10 CFR 50.55a(a)(3)(ii) for the licensee's fourth 10-year ISI interval. All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions concerning this matter, please contact Dr. Sreenivas, at (301) 415-2597.

Sincerely,

Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-339

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 (ISI) INTERVAL

RELIEF REQUESTS NOS. SPT-001, SPT-002, SPT-003, SPT-004, SPT-005, AND SPT-006

NORTH ANNA POWER STATION, UNIT NO. 2

VIRGINA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

1.0 INTRODUCTION

By letter dated February 1, 2010, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100330125), Virginia Electric and Power Company (the licensee), requested relief from certain requirements of the American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code* (ASME Code), 2004 Edition, under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a(g)(4), for the fourth 10-year ISI Program for North Anna Power Station (NAPS), Unit No. 2. The NAPS, Unit No. 2, fourth 10-year ISI interval is scheduled to start on December 14, 2010, and end on December 13, 2020.

The U.S. Nuclear Regulatory Commission (NRC) staff has concluded based on the information provided by the licensee, that pursuant to 10 CFR 50.55a(a)(3)(ii), Relief Requests (RRs) SPT-001 through SPT-006 are authorized on the basis that compliance with the specified requirements of the ASME Code would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

Section 50.55a(g) requires that 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). 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 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) twelve 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 ISI of NAPS, Unit No. 2, is the 2004 Edition of the ASME Code, Section XI.

3.0 TECHNICAL EVALUATION

3.1 Relief Request 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 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

> 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, Unit No. 2, 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 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 due to reactor coolant at temperature above 350 degree F and limited air circulation in the vessel cubicle.

In addition, the examiner is limited to approximately a 30-minute supply of oxygen in the breathing apparatus for the containment entry/exit including the examination.

(e) Licensee's Proposed Alternative

> The improved technical specification (ITS) establishes 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 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 that RCS operational leakage is within limits by performance of a RCS inventory balance at a frequency not exceeding 72 hours. The ITS also require that 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. Additionally, 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.

These actions would identify any integrity concerns associated with this area.

(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, Unit No. 2, pertaining to VT-2 visual 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 RCS. Since the containment building is at sub-atmospheric condition during the system leakage test, the examiner must wear 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 to the 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 ITS require monitoring of the reactor coolant leak rate, atmospheric particulate radioactivity, and containment sump level. The NRC staff evaluated 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 finds that the licensee's proposed alternative will provide reasonable assurance that unallowable inservice leaks, if developed at the bottom of 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 the system leakage test 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 Code-required VT-2 visual examination of the bottom of the reactor vessel during the system leakage test 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 SPT-001, is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the fourth 10-year ISI interval of NAPS, Unit No. 2. 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.

3.2 Relief Request SPT-002

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

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 Number B15.10, requires that a system leakage test be performed each refueling outage in accordance with the requirements of IWB-5220. The 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 the interval system leakage test with the isolation valves in the normally closed position which corresponds to the RCS at nominal operating pressure and at near operating temperature.

(d) Licensee's Basis for Requesting Relief

The subject piping segments are equipped with valves that provide for double-isolation of the RCS pressure boundary (RCPB). 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 stated in the application:

> 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 these 1 inch or less in diameter RCPB vent / drain / test / sampling /instrument 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.
>
> The non-isolable portion of the RCPB 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 reactor coolant system (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 integrity of the RCS boundary.

(e) Licensee's Proposed Alternative

As an alternative to the Code-required pressure test of the subject Class 1 RCS pressure boundary connections, the following is proposed:

> The RCPB vent, drain, instrumentation, test, and sample 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)].
>
> The RCPB vent, drain, instrumentation, test, and sample 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(b)]. This examination will be performed with the RCS at nominal operating pressure and at near operating temperature.

(f) NRC Staff's Evaluation

The ASME Code, Section XI Code of Record 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 SPT-002, the licensee proposed an alternative to the 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 RCS. The line configuration, as outlined, provides double-isolation of the RCS. Under normal plant operating conditions, the subject pipe segments would see 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. 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 finds 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 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

SPT-002 is authorized for the fourth 10-year intervals of NAPS, Unit No. 2. 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.

3.3 Relief Request SPT-003

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

Residual heat removal (RHR) system components and piping between 2-RH-MOV-2700 and 2-RH-MOV-2701 (RHR suction) in drawing 12050-CBM-094A-4, sheet 1 of 2.

(b) ASME Code Requirements

Table IWB-2500-1, Item Number B15.10 (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% 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 RHR piping between 2-RH-MOV-2700 and 2-RH-MOV-2701 (RHR suction) including the valves during the Class 2 system pressure test.

(d) Licensee's Basis for Requesting Relief

> Normal reactor coolant pressure at 100% rated power is approximately 2235 psig. The piping in question is separated from this reactor coolant pressure by a single closed valve, and as such does not normally see this pressure. Opening of valve 2-RH-MOV-2700 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 Code of Record, 2004 ASME Code Section XI, Table IWB-2500-1, Category B-P, Item number B15.10 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 normal operating pressure of the RCS corresponding to 100% rated reactor power and shall include all Class 1 components within the RCS boundary. In RR SPT-003, the licensee proposed an alternative to the boundary subject to test pressurization required under the Code of Record 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 see RCS temperature and pressure only if leakage through an inboard isolation valve occurs. As requested in RR 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. 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.

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 concurs with the licensee's finding that compliance with the ASME Code requirement would result in hardship without a compensating increase in the level of quality and safety. The licensee, however, 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 proposal 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 finds 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) <u>Conclusion</u>

It is concluded that test pressurization during a 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 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 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 RR SPT-003 is authorized for the fourth 10-year ISI interval of NAPS, Unit No. 2. 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 SPT-004

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

½" to 10" NPS Safety Injection (SI) piping associate valves
Segment Boundary: (valve-to-valve) from 2-SI-91, 2-SI-99, and 2-SI-105, to 2-SI-MOV-2890C and 2-SI-MOV-2890D in drawings 12050-CBM-096B-4, sheet 4 of 4, and 12050-CBM-096A-4, sheet 2 of 3 (Low Head Safety Injection to the Reactor Coolant Cold Legs)

and

¾" to 10" NPS Safety Injection piping associate valves
Segment Boundary: (valve-to-valve) from 2-SI-112, 2-SI-117, and 2-SI-124, to 2 -SI-MOV-2890A and 2-SI-MOV-2890B, in drawings 12050-CBM-096B-4, sheet 4 of 4 and 12050-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, Item Number B15.10 (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% 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% rated power is approximately 2235 psig. The piping in question is separated from this reactor coolant pressure by check valves, and as such does not normally see 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% reactor power. However, there is no provision to pressurize the segment of piping between the check valves to the 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 in question 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 VT-2 visually examined 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 Code of Record, 2004 Edition, ASME Code, Section XI, Table IWB-2500-1, Category B-P, Item number B15.10, 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 RCS corresponding to 100% rated reactor power and shall include all Class 1 components within the RCS boundary. In RR SPT-004, the licensee proposed an alternative to the boundary subject to test pressurization required under the Code of Record 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 subject pipe segment would see RCS temperature and pressure only if leakage through an inboard isolation valve occurs. As requested in RR 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 a 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 concurs with the licensee's finding that compliance with the Code requirement would result in hardship without a compensating increase in the level of quality and safety. The licensee, however, 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 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 finds 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) Conclusion

It is concluded that test pressurization during a system leakage test of the Class 1 pressure retaining components within the system boundary of the SI system between an inboard and an outboard isolation valve including the valves as required by the 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 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 RR SPT-004 is authorized for the fourth 10-year ISI interval of NAPS, Unit No. 2. 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 Relief Request SPT-005

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

Reactor Coolant System

Components: Class 1 components and piping between 2-RC-R-1 (Reactor Inner O-ring), 2-RC-32 and 2-RC-HCV-2544 in drawing 12050-CBM-093A-4, sheet 1 of 3

(b) ASME Code Requirements

Table IWB-2500-1, Item No. B15.10 (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% 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

> Normal reactor coolant pressure at 100% 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

The Code of Record, 2004 Edition, ASME Code, Section XI, requires that all Class 1 components within the RCS boundary undergo a system leakage test during each refueling outage. In RR SPT-005, the licensee requested relief from performing a system leakage test of the reactor vessel head flange seal leak detection piping at the Code-required test pressure corresponding to the nominal operating pressure during system operation. 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 configuration of this line makes the Code-required system leakage test difficult either with the vessel head in place or removed. The piping cannot be filled completely with water since it cannot 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.

If the licensee were to perform the system leakage test in accordance with the Code requirement by pressurizing the space between the inner and the outer O-ring seals, it will likely fail the inner O-ring and subsequently require replacement of the damaged O-ring with a new O-ring. This will result in loss of outage time and at the same time expose test crew to additional radiation in the process of de-tensioning and removal of the reactor vessel head, replacement of the inner O-ring and the installation of the reactor vessel head. This evolution would create extreme hardship to the licensee without a compensating increase in the level of quality and safety. The licensee, however, has proposed to perform a VT-2 visual examination of the reactor vessel head flange seal leak detection piping during a refueling outage. The NRC staff evaluated that the hydrostatic head developed due to water above the vessel flange during flood-up. This will allow for the detection of any gross inservice flaws if present in the subject piping. The proposed testing would provide reasonable assurance of structural integrity. Therefore, it is acceptable.

(g) Conclusion

Based on NRC staff's evaluation, a system leakage test of the reactor vessel head flange seal leak detection piping at the Code-required test pressure corresponding to the nominal operating pressure during system operation would cause hardship to the licensee without a compensating increase in the level of quality and safety. The licensee's proposed alternative provides reasonable assurance of structural integrity. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative in RR SPT-005 is authorized for the fourth 10-year ISI interval of NAPS, Unit No. 2. 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.

3.6 Relief Request SPT-006

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

Chemical and Volume Control System

Components: NPS 2" Auxiliary Spray Piping
Segment Boundary (valve-to-valve) from 2-CH-HCV-2311 to 2-CH-341 in drawing 12050-CBM-095C-4, sheet 1 of 2

(b) ASME Code Requirements

> IWB-5222(b) states, "The pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval shall extend to all Class 1 pressure retaining components within the system boundary." IWB-5221(a) states, "The system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power." The paragraphs require pressurization at RCS nominal operating pressure on this portion of the extended Class 1 boundary at or near the end of the interval.

(c) Licensee's Request for Relief

For Class 1 auxiliary pressurizer spray line between isolation valves 2-CH-HCV-2311 and 2-CH-341, a system leakage test shall be conducted at or near the end of each inspection interval prior to reactor startup. The segment of Class 1 piping including the valves in the system boundary will be visually examined for evidence of past leakage and/or leakage during the system leakage test performed at nominal operating pressure associated with 100% reactor power.

(d) Licensee's Basis for Requesting Relief

> The pressurizer pressure is maintained by normal pressurizer spray which uses the Reactor Coolant Pumps (RCP). Normal pressurizer spray is controlled by the Pressurizer Pressure Control System which automatically controls the pressurizer environment. The primary purpose of the auxiliary spray line is for pressure control when the RCPs are not running (i.e., during a post accident condition when it is desired to decrease the Reactor

Coolant System (RCS) pressure). Operation of the auxiliary spray line at hot standby or at power would lead to an unnecessary plant transient. In order to meet the ASME Code requirement, the normally closed upstream isolation 2-CH-HCV-2311 must be opened to pressurize the subject piping segment. The charging pumps take suction from the volume control tank and discharge to the pressurizer at a slightly higher pressure than that of the RCS. Therefore, opening of the valve 2-CH-HCV-2311 at hot standby or at power would increase pressurizer spray flow which will cause an adverse reduction in RCS pressure. Further, with the piping segment being at the containment ambient temperature and the RCS at its nominal operating temperature, any injection of cold water into the pressurizer would cause a thermal shock in the spray piping and the spray nozzle.

(e) Licensee's Proposed Alternative

Testing of this piping segment at RCS operating pressure does not provide a compensating increase in the level of quality and safety for the following reasons:

1. The design pressure rating of this piping segment is the same as the RCPB; however, the operating pressure of the piping segment is well below the normal RCS operating pressure.

2. This segment is isolated from the RCS pressure under normal operating condition.

3. This segment is subject to ASME Code required VT-2 visual examination. This examination is performed with the segment isolated from the RCS and the RCS at its normal operating pressure and temperature. This examination is performed each refueling outage and is sufficient to identify any structural defect that could potentially challenge the integrity of the segment during normal operation.

(f) NRC Staff's Evaluation

The Code of Record, 2004 Edition of the ASME Code, Section XI, Table IWB-2500-1, Category B-P, Item number B15.10, requires that a system leakage test of Class 1 pressure retaining piping and valves be performed prior to plant startup following each reactor refueling outage. This leakage test must be performed at a system pressure not less than the pressure corresponding to 100% rated reactor power. Paragraph IWB-5222(b) of the Code on "Boundaries" states:

> The pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval shall extend to all Class 1 pressure retaining components within the system boundary.

The subject portion of the auxiliary spray piping between isolation valves 2-CH-341 and 2-CH-HCV-2311 corresponds to the reactor coolant pressure boundary and, is required to be pressure tested in accordance with Paragraph IWB-5222(b). However, if auxiliary spray valve 2-CH-HCV-2311 is opened while the RCS is at normal pressure and temperature, cold water would be injected into the pressurizer unnecessarily, causing a drop in RCS pressure. This would cause an off-normal plant transient and create hardship on the licensee.

The subject piping segment at NAPS, Unit No. 2, is 2" nominal pipe size, Schedule 160, stainless steel pipe between motor control valve 2-CH-HCV-2311 and check valve 2-CH-341 that connects the auxiliary spray line to the normal pressurizer spray line. In the normal operating mode, the pressurizer spray line (downstream of check valve 2-CH-341) is pressurized by the RCPs. Piping in the auxiliary spray system (upstream of 2-CH-HCV-2311) is pressurized by the charging pumps. Thus, the line segment between these valves would be pressurized if the auxiliary system is activated, which would only occur when the RCPs are not running (i.e., during a post-accident condition when it is desired to decrease RCS pressure). To require the licensee to activate auxiliary spray for the purpose of system leakage test with RCS at temperature and pressure would cause a thermal shock transient in the pressurizer spray piping and the spray nozzle.

The licensee has proposed to conduct the system leakage test for this piping segment in accordance with Paragraph IWB-5222(a) of the Code which states:

> The pressure retaining boundary during the system leakage test shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation startup. The visual examination shall, however, extend to and include the second closed valve at the boundary extremity.

The licensee, however, has proposed an alternative to visually examine (VT-2) for leaks in the isolated portion of the subject segments of piping with the isolation valves in the normal reactor operation startup which would indicate any evidence of past leakage during the operating cycle as well as any active leakage during the system leakage test. The NRC staff has determined that the licensee's proposed alternative would ensure leakage integrity of the subject piping segment and would meet the intent of the Code requirement.

Further, pressurization of the piping segment in the auxiliary spray line between the isolation valves 2-CH-HCV-2311 and 2-CH-341 to meet the Code requirement on a system leakage test, the RCS may subject to an off-normal transient which may have an adverse impact on RCS components. This may cause a hardship to the licensee without a compensating increase in the level of quality and safety.

(g) <u>Conclusion</u>

It is concluded that test pressurization during a system leakage test of the Class 1 pressure retaining components within the system boundary of pressurizer auxiliary spray line in RR SPT-006 as required by the 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 the subject RR provides a reasonable assurance of structural integrity for the subject piping segments. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative in RR SPT-006 is authorized for the fourth 10-year ISI interval of NAPS, Unit No. 2. 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.

4.0 <u>CONCLUSION</u>

As set forth above, the NRC staff has determined that the compliance with the ASME Code requirements would result in a hardship or unusual difficulty without a compensating increase in quality or safety. Furthermore, the NRC staff concludes that the licensee's proposed alternative provide reasonable assurance of structural integrity. Therefore, the licensee's proposed alternatives for RRs SPT-001 through SPT-006 in lieu of the required ASME Code is authorized for the fourth 10-year interval, pursuant to 10 CFR 50.55a(a)(3)(ii).

All other ASME Code 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.

Therefore, the NRC staff authorizes the alternatives and relief noted above, at NAPS, Unit No. 2, for the fourth 10-year ISI interval, which starts on December 14, 2010, and ends on December 13, 2020.

Principal Contributor: Pat Patnaik, NRR

Date: November 2, 2010

If you have any questions concerning this matter, please contact Dr. Sreenivas, at (301) 415-2597.

Sincerely,

/RA/

Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-339

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