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

May 7, 2013

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
Post Office Box 1295 / Bin 038
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 (VEGP) - SAFETY EVALUATION OF RELIEF REQUEST VEGP-ISI-ALT-09, VERSION 2.0, REGARDING LEAKAGE TESTS FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS (TAC NOS. MF0673 AND MF0674)

Dear Mr. Pierce:

By letter to the U.S. Nuclear Regulatory Commission (NRC), dated March 6, 2013, as supplemented by letter dated March 13, 2013, Southern Nuclear Operating Company, Inc. (SNC, the licensee) requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code to conduct an alternative (VEGP-ISI-ALT-09, VERSION 2) to the required leakage tests for reactor coolant pressure boundary components.

On March 15, 2013, the NRC staff verbally authorized the licensee's proposed alternative for four valves located in Vogtle, Unit 2, which are scheduled for inspection in the spring 2013 refueling outage. This safety evaluation provides the basis for that verbal authorization. This safety evaluation also provides the basis for the staff's position regarding the remainder of the licensee's request for relief which was not considered in the verbal authorization.

The NRC staff determines that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the subject components and that complying with the specified requirement would result in a hardship or unusual difficulty without a compensating increase in the 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(a)(3)(ii). Therefore, in accordance with 10 CFR 50.55a(a)(3)(ii) the NRC staff authorizes the proposed alternative at Vogtle Electric Generating Plant, Units 1 and 2, for a period of time not to extend beyond the ends of refueling outages 1 R19 (fall 2015) and 2R18 (spring 2016) for Units 1 and 2, respectively.

All other requirements in ASME Code, Section XI, and 10 CFR 50.55a for which relief was not specifically requested and approved in this RR remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Sincerely,

R Pascarelli

Robert Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-424 and 50-425

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST VEGP-ISI-ALT-09, VERSION 2

REQUEST FOR RELIEF FROM ASME BOILER AND PRESSURE VESSEL CODE, SECTION XI

REQUIREMENTS FOR CVCS THREE-INCH CLASS 1 RCPB LEAKAGE TEST

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

DOCKET NO. 50-424 AND 50-425

1.0 INTRODUCTION

By letter dated March 6, 2013 (Agencywide Document Access and Management System (ADAMS) Accession No. ML13066A334) as supplemented by letter dated March 13, 2013 (Accession No. ML13073A089), Southern Nuclear Operating Company, Inc. (the licensee), requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code to conduct an alternative to the required leakage tests for reactor coolant pressure boundary components.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii) the licensee proposed an alternative to article IWA-5241(a) of Section XI of the ASME Boiler and Pressure Vessel Code to permit examination at the boundaries of seal cap enclosures as well as examination of bolting in 8 three-inch reactor coolant pressure boundary valves in lieu of the required leakage test 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.

On March 15, 2013, the NRC staff verbally authorized the licensee's proposed alternative for four valves located in Vogtle, Unit 2, which are scheduled for inspection in the spring 2013 refueling outage. This safety evaluation provides the basis for that verbal authorization. This safety evaluation also provides the basis for the staff's position regarding the remainder of the licensee's request for relief which was not considered in the verbal authorization.

2.0 REGULATORY EVALUATION

In this relief request the licensee requests authorization of an alternative to the requirements of article IWA-5241(a) of Section XI of the ASME Code pursuant to 10 CFR 50.55a(a)(3)(ii).

Enclosure

Adherence to article IWA-5241(a) of Section XI of the ASME Code is mandated by 10 CFR 50.55a(g)(4) which states, in part, that throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions and addenda of the ASME Boiler and Pressure Vessel Code.

10 CFR 50.55a(a)(3) states, in part, that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used, when authorized by the NRC, if the licensee demonstrates (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above analysis, the staff finds that regulatory authority to authorize an alternative to article IWA-5241(a) of Section XI of the ASME Code, as requested by the licensee, exists.

3.0 TECHNICAL EVALUATION

3.1 Applicable Code Edition and Addenda

ASME Boiler and Pressure Vessel Code, Section XI, 2001 Edition, 2003 Addenda

3.2 Components for Which Relief is Requested

Components: 8 3-inch reactor coolant pressure boundary check valves in the chemical and volume control system. Four valves are located in each unit (unit 1 valve numbers: 11208-U6-035, 11208-U6-036, 11208-U6-037, and 11208-U6-038; Unit 2 valve numbers: 21208-U6-035, 21208-U6-036, 21208-U6-037, and 21208-U6-038)

Code Class Class 1

3.3 Reason for Request

In its request, the licensee states that the valves under consideration require inspection under ASME Code Section XI Table IWB2500-1, Examination Category B-P, Item B15.10. This item requires that a system leakage test be conducted prior to plant startup following each reactor refueling outage in accordance with ASME Code Section XI Article IWB 5220 through the use of a VT-2 examination in accordance with ASME Code Section XI Article IWA 5240.

In its request, the licensee also states that the valves under consideration are equipped with seal cap enclosures to mitigate leakage of borated water past the body-to-bonnet gasket. The staff notes that, while the seal cap enclosures are designed to retain leakage, they do not function as the system pressure boundary. That function is retained by the body-to-bonnet bolts. The licensee stated that, in addition to containing gasket leakage, the seal cap enclosures are associated with two recently identified concerns; first, the seal cap enclosures prevent inspection of the pressure boundary bolts as required by Section XI of the ASME Code; and second, leakage into the seal cap enclosure may create an environment in which the bolting material is subject to failure due to stress corrosion cracking. The licensee stated that

the first issue was the basis for the request; however, the second issue was significant in developing the proposed alternative.

In support of the need for the requested relief, the licensee stated that the immediate replacement of the valves under consideration was not possible due to the lead time for obtaining replacement valves. Given that immediate replacement of the valves was not possible, the licensee stated that in order to satisfy the ASME Code requirement with the current valve design and configuration, the seal caps would need to be removed and remain removed during the required leakage test. Subsequently, the seal cap enclosures could remain uninstalled, could be reinstalled at mode 3 conditions, or the plant could be returned to mode 5 for reinstallation of the seal cap enclosure.

In analyzing the available options, the licensee indicated that it would be highly undesirable to operate with the seal caps removed as this could result in uncontained reactor coolant leakage. Uncontained leakage from these valves could require repair of the body to bonnet gasket. Such a repair would require that the plant be shut down and drained to mid loop. The licensee found reinstallation of the seal cap enclosures at mode 3 to be undesirable due to the fact that substantial work (e.g., seal cap reinstallation, insulation reinstallation, and scaffold removal) would need to be performed under conditions, which are detrimental to personnel safety, e.g., heat stress. The licensee acknowledge that reinstallation of the seal cap enclosures at mode 5 would resolve most of the personnel issues inherent to reinstallation of the seal cap enclosures at mode 3 but would result in an undesirable delay in the refueling outage and impose additional thermal cycling of primary plant components.

3.4 Proposed Alternative

In its request, as an alternative to the leakage test required by ASME Code Section XI Table IWB2500-1, the licensee proposed to perform a leakage test meeting all the conditions and requirements of the required leakage test excepting that the proposed leakage test would be conducted with the seal cap enclosures installed and the presence or absence of leakage would be identified at the seal cap enclosure welds rather than at the valve's actual pressure boundary. The licensee also stated that such examinations will be performed following the conclusion of each refueling outage at Plant Vogtle, Units 1 and 2, until each of the seal cap enclosures are permanently removed.

In addition to the leakage test proposed above, the licensee proposed that, during the upcoming 2R16 (spring 2013) refueling outage, it will remove the seal cap enclosure from 2-1208-U6-037, and perform a VT-3 and a UT on the valve's body-to-bonnet bolting. Additionally, the licensee proposed to verify the absence of leakage of reactor coolant into the remaining Unit 2 valves equipped with seal cap enclosures (2-1208-U6-035, 2-1208-U6-036, and 2-1208-U6038) by creating an orifice into the seal cap enclosures and examining the interior of the enclosures. If evidence of leakage is detected, the licensee proposed to remove the seal cap enclosure(s) from the affected valve(s) and to examine the bolting using VT-3 and UT.

In subsequent refueling outages the licensee proposed to repair, replace, or modify the valves equipped with seal cap enclosures to permit the permanent removal of the enclosures or to examine the enclosures in accordance with Table 1 of its request. The proposed examinations consist of removing the seal cap enclosures and performing a VT-3 exam on all bolts. Additionally, a UT exam will be performed on all bolts which have not been previously examined

by UT or where evidence of leakage of reactor coolant into the seal cap enclosure exists. The licensee has committed to modify or replace the valve bonnets with an enhanced design not later than 1 R19 (fall 2015) and 2R18 (spring 2016) for Units 1 and 2, respectively. The licensee indicated that the enhanced bonnet design would make the pressure boundary of the valve accessible for inspection and would preclude exposure of the pressure boundary bolting to environments which may cause stress corrosion cracking.

3.5 Licensee's Technical Basis

In its request, the licensee provides the following information in support of its request to conduct an alternative to the required leakage tests for reactor coolant pressure boundary components for a period of time not to exceed the ends of refueling outages 1 R19 (fall 2015) and 2R18 (spring 2016) for Units 1 and 2, respectively.

a. The conduct of a leakage test at the boundary of the seal cap enclosure, as opposed to the pressure boundary, is sufficient to identify leakage from the reactor coolant system into containment.

b. The inspection program proposed exceeds industry guidelines provided in PWROG Letter OG-12-330.

c. The scope and frequency of examination provided for in PWROG Letter OG-12-330 provide reasonable and sufficient assurance that the valve bolting continues to perform its intended function.

3.6 Staff Evaluation

As previously stated in the Regulatory Evaluation section (2.0) of this safety evaluation, prior to authorizing the proposed alternative under 10 CFR 50.55a(a)(3)(ii), the staff must find that the technical information provided in support of the proposed alternative is sufficient to demonstrate that compliance with ASME Code Section XI, Article IWA-5241(a): a) would result in a hardship or unusual difficulty; and, b) would not provide a compensating increase in the level of quality and safety when compared to the proposed alternative. If these criteria are met, the staff finds that the proposed alternatives to ASME Code requirements will provide reasonable assurance of structural integrity or leak tightness of the subject components.

The staff has reviewed the process, as described by the licensee, required to complete a code compliant repair to the component under consideration. The staff agrees that the options presented for meeting the code requirements are complete, i.e., no other reasonable options are available. The staff also agrees that:

a. operation without the seal cap enclosures in place would be undesirable due to the potential for leakage;

b. replacement of the seal cap enclosures under mode 3 conditions would involve personnel hazards; and

c. returning the plant to mode 5 to replace the seal cap enclosures would extent outages and, more importantly, would subject the plant to unnecessary pressure cycles which could lead to issues associated with low cycle fatigue

Based on the above, the staff agrees with the licensee's contention that conducting the required inspection constitutes a hardship. This satisfies the first condition of 10 CFR 50.55a(a)(3)(ii).

In considering the second condition of 10 CFR 50.55a(a)(3)(ii), whether adherence to the ASME Code requirement would provide an increase in quality and safety commensurate with the hardship or unusual difficulty imposed by meeting the code requirement, the staff evaluated the proposed alternative as well as the technical basis for the alternative as proposed by the licensee and described in Section 3.4 and 3.5, above. The staff finds no reason to object to the technical accuracy and/or sufficiency of any of the information provided.

In reviewing the licensee's proposal, the staff considered two issues. First, the staff considered the potential for and consequences of leakage of reactor coolant from the reactor coolant system into containment and second, the staff considered the structural integrity of the valve pressure boundary, i.e., the structural integrity of the valve body to bonnet bolts. In considering these two issues the staff noted:

a. While the seal cap enclosures are not designed to function as the valve pressure boundary, if that pressure boundary remains intact, the seal cap enclosures are designed to contain reactor coolant leakage through the body to bonnet gasket and prevent the coolant from leaking into containment.

b. The pressure boundary bolting is constructed from type A286 stainless steel. This material is subject to intergranular stress corrosion cracking in hot, high purity water environments containing some oxygen. An environment meeting these conditions may be present in the enclosures as a result of leakage of primary coolant past the valve body to bonnet gasket. Conversely, when not exposed to the above environment, A286 stainless steel is a highly effective material for use as pressure boundary bolting.

c. Intergranular cracks in type A286 stainless steels grow slowly in the environment described in item b.

Relative to items a – c, above, the staff finds that the critical elements of the licensee's proposed alternative are:

a. A visual inspection of the seal cap enclosure boundary will be conducted at the end of each refueling outage.

b. During the spring 2013 refueling outage, seal cap enclosures will be remove for VT-3 and UT examination of the bolts or the inside of the seal cap enclosure will be examined for evidence of past or present leakage of reactor coolant into the seal cap enclosure. If leakage is present the seal cap enclosure will be removed and the bolts examined by VT-3 and UT.

c. During subsequent refueling outages all seal cap enclosures will be removed. Bolts will be examined by VT-3. Bolts will also be examined by UT if 1) they have not previously

been examined by UT or 2) evidence of leakage of reactor coolant into the seal cap enclosure is identified.

The staff finds that the licensee's inspection program will be effective in preventing both loss of structural integrity of the pressure boundary bolting and preventing leakage into containment because:

a. The seal cap enclosures are examined once per refueling outage by either removing the seal cap enclosure or internally examining the seal cap enclosure for leakage. In the absence of leakage cracking of the pressure boundary bolts and, therefore, loss of pressure boundary integrity is not of concern. Based on the continued integrity of the valve pressure boundary, rapid failure of the seal cap enclosure is not of concern. In the absence of a credible threat of rapid seal cap enclosure failure, an inspection of the seal cap enclosure boundary for leakage provides reasonable assurance that reactor coolant will not leak into containment.

b. In the event that reactor coolant is found in any seal cap enclosure, body-to-bonnet bolts are examined by VT-3 and UT. These examinations will identify any existing cracks. If cracks are located, repairs will be made. Based on the slow growth rate of intergranular cracks in type A286 stainless steel under the environmental conditions which may be found in the seal cap enclosures, it is reasonable to conclude that, in the absence of an identified crack, a bolt will not fail due to cracking prior to the next scheduled examination. Based on the maintained integrity of the pressure boundary between examinations, the proposed leak test provides reasonable assurance that reactor coolant will not leak into containment as described above.

Based on the above analysis, the staff finds that the licensee has provided sufficient information to provide reasonable assurance that, by employing the proposed alternative, the structural integrity of the valves and the leak tight integrity of the reactor coolant system can be maintained during the time period for which relief has been requested.

Given that the licensee has provided sufficient information to the staff to provide reasonable assurance of the structural and leak tight integrity of the component during the period for which relief has been requested, the staff finds that adherence to the ASME code requirement does not provide a compensating increase in the level of quality and safety when compared to the proposed alternative. The second criterion in 10 CFR 50.55a(a)(3)(ii) is, therefore, met.

Based on the above analysis, the staff finds that the technical requirements of 10 CFR 50.55a(a)(3)(ii) have been met and, therefore, that the licensee's proposal provides reasonable assurance of structural and leak tight integrity of the subject components. The staff, therefore, finds no technical basis that would preclude it from authorizing an alternative to Article IWA-5241(a), of Section XI, of the ASME Code as requested by the licensee.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative provides reasonable assurance of structural integrity or leak tightness of the subject components and that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10CFR 50.55a(a)(3)(ii). Therefore, the NRC staff authorizes the proposed alternative at Vogtle Electric Generating Plant, Units 1 and 2, for a period of time not to extend beyond the ends of refueling outages 1 R19 (fall 2015) and 2R18 (spring 2016) for Units 1 and 2, respectively.

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

Principal Contributor: David Alley

Date: May 7, 2013

All other requirements in ASME Code, Section XI, and 10 CFR 50.55a for which relief was not specifically requested and approved in this RR remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Sincerely,

/RA/

Robert Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-424 and 50-425

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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