Charles R. Pierce
Regulatory Affairs Director

Southern Nuclear
Operating Company, Inc.
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, AL 35242

Tel 205.992.7872
Fax 205.992.7601

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Docket Nos.: 50-424
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NL-16-0724

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Vogtle Electric Generating Plant, Units 1 & 2
Proposed Inservice Inspection Alternative VEGP-ISI-ALT-11, Version 1.0

Ladies and Gentlemen:

In accordance with 10 CFR 50.55a(z)(1), Southern Nuclear Operating Company (SNC) hereby requests Nuclear Regulatory Commission (NRC) approval of proposed inservice inspection (ISI) alternative VEGP-ISI-ALT-11, Version 1.0. This alternative requests an elimination of the Reactor Pressure Vessel Threads in Flange examination requirement.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at (205) 992-7369.

Sincerely,

C. R. Pierce

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Regulatory Affairs Director

CRP/JMC/lac

Enclosure: Proposed Alternative VEGP-ISI-ALT-11, Version 1.0, in Accordance with 10 CFR 50.55a(z)(1)

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. D. R. Madison, Vice President – Fleet Operations
Mr. M. D. Meier, Vice President – Regulatory Affairs
Mr. B. K. Taber, Vice President – Vogtle 1 & 2
Mr. B. J. Adams, Vice President – Engineering
Mr. G.W. Gunn, Regulatory Affairs Manager – Vogtle 1 & 2
RType: CVC7000

U. S. Nuclear Regulatory Commission
Ms. C. Haney, Regional Administrator
Mr. R. E. Martin, NRR Senior Project Manager – Vogtle 1 & 2
Mr. T. Stephen, Senior Resident Inspector – Vogtle 1 & 2

Vogtle Electric Generating Plant, Units 1 & 2
Proposed Inservice Inspection Alternative VEGP-ISI-ALT-11, Version 1.0

Enclosure

Proposed Alternative VEGP-ISI-ALT-11, Version 1.0,
in Accordance with 10 CFR 50.55a(z)(1)

Enclosure to NL-16-0724

Proposed Alternative VEGP-ISI-ALT-11, Version 1.0, in Accordance with 10 CFR 50.55a(z)(1)

Plant Site-Unit:

Vogtle Electric Generating Plant (VEGP) - Units 1 and 2

Interval-Interval Dates:

3rd Inservice Inspection (ISI) Interval, May 31, 2007 through May 30, 2017

Requested Date for Approval and Basis:

Approval is requested by February 28, 2017 to allow implementation of the proposed alternative during the 20th Refueling Outage of Vogtle Unit 1. This proposed alternative would also apply to all the remaining refueling outages for both units during the third ISI Interval.

ASME Code Components Affected:

ASME Category B-G-1, Item B6.40, Pressure retaining bolting greater than 2-inches, Reactor Vessel – Threads in Flange require a volumetric examination.

Applicable Code Edition and Addenda:

The applicable Code edition and addenda is ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition with Addenda through 2003.

Applicable Code Requirements:

The Reactor Pressure Vessel (RPV) threads in flange are examined using a volumetric examination technique with 100% of the flange ligament areas examined every ISI interval. The examination area is the one-inch area around each RPV stud hole, as shown on Figure IWB-2500-12.

Background and Reason for Request:

Southern Nuclear Operating Company (SNC) has worked with the industry to evaluate eliminating the RPV Threads in Flange examination requirement. Licensees in the US and internationally have worked with the Electric Power Research Institute (EPRI) to produce a technical report (Reference 1) which provides the basis for elimination of the requirement. The report includes a survey of inspection results from over 168 units, a review of operating experience related to RPV flange/bolting, a flaw tolerance evaluation and a bounding assessment on risk if these examination requirements are eliminated. The conclusion from this evaluation is that the elimination of the current examination results in a negligible increase in risk which is not commensurate with the associated burden (worker exposure, personnel safety, radwaste, critical path time) of the examination. The technical basis for this alternative is discussed in more detail below.

An evaluation of potential degradation mechanisms that could impact flange/threads reliability was also performed. Potential types of degradation evaluated included: pitting, intergranular attack, corrosion fatigue, stress corrosion cracking, crevice corrosion, velocity phenome, dealloying corrosion, general corrosion, stress relaxation, creep, mechanical wear and mechanical/thermal fatigue. Other than the potential for mechanical/thermal fatigue, there are no active degradation mechanisms identified for the threads in flange component. The EPRI report documents a stress analysis and flaw tolerance evaluation of the flange thread area to assess mechanical/thermal fatigue potential. The Pressurized Water Reactor (PWR) design was selected because of its higher design pressure and temperature. A representative geometry for the finite element model used the largest PWR RPV diameter along with the largest bolts and the highest number of bolts. The larger and more numerous bolt configuration results in less flange material between bolt holes, whereas the larger RPV diameter results in higher pressure and thermal stresses.

The resulting crack growth calculated was negligible due to the small stress intensity variation (delta K) and the relatively low number of cycles associated with the transients evaluated. Because the crack growth is insignificant, the allowable flaw size will not be reached and the integrity of the component is not challenged for at least 80 years (original 40-year design life plus additional 40 years of plant life extension), indicating that the component is very flaw tolerant. This clearly demonstrates that the Threads in Flange examinations can be eliminated without affecting the safety of the RPV.

As discussed above, the results of the survey confirmed that the RPV Threads in Flange examination are adversely impacting outage activities (dose, safety, critical path time) while not identifying any service induced degradations. Specifically, for the US fleet, a total of 94 units have responded to date and none of these units have identified any type of degradation. As can be seen in Table 1 below, the data is encompassing. The 94 units represent data from 33 Boiling Water Reactor (BWRs) and 61 PWRs. For the BWR units, a total 3,793 examinations were conducted and for the PWR units a total of 6,869 examinations were conducted, with no service induced degradation identified. The response data includes information from all of the plant designs in operation in the US and includes BWR-2, -3, -4, -5 and -6 designs. The PWR plants include the 2- loop, 3-loop and 4-loop designs and each of the PWR NSSS designs (i.e., Babcock & Wilcox, Combustion Engineering and Westinghouse).

Table 1: Summary of Survey Results – US Fleet

Plant Type	Number of Units	Number of Examinations	Number of Reportable Indications
BWR	33	3,793	0
PWR	61	6,869	0
Total	94	10,662	0

In addition to the examination history and flaw tolerance discussed above, the EPRI report discusses studies conducted in response to the issuance of the Anticipated Transient Without Scram (ATWS) Rule by the United States Nuclear Regulatory Commission (USNRC). This rule was issued to require design changes to reduce expected ATWS frequency and consequences. Many studies have been conducted to understand the ATWS phenomena and key contributors to successful response to ATWS event. In particular, the reactor coolant system and its individual components were reviewed to determine weak links. As an example, even though significant structural margin was identified in USNRC SECY-83-293 for PWRs, the ASME Service Level C pressure of 3200 psig was assumed be to an unacceptable plant condition. While a higher ASME service level might be defensible for major Reactor Coolant System (RCS) components, other portions of the RCS could deform to the point of inoperability. Additionally, there was the concern that steam generator tubes might fail before other RCS components, with a resultant bypass of containment. The key take away for these studies is that the RPV flange ligament was not identified as a weak link and other RCS components were significantly more limiting. Thus, there is substantial structural margin associated with the RPV flange.

EPRI determined that conducting a plant-specific quantitative risk impact analysis was not warranted based on the bounding analysis provided in their report. Conducting a plant-specific quantitative risk impact analysis would require additional unnecessary resources and would not add any new risk insights. To have an impact on risk, some measurable failure potential must exist (i.e., some type of operative degradation), conditions that could create a credible consequence must exist (e.g., cause a plant initiating event, impact the plant's mitigative ability, or some combination of both) and the end result needs to have a measurable impact on plant risk.

The EPRI report identifies that the RPV Threads in Flange are performing with very high reliability based on operating and examination experience. This is due to the robust design and a relatively benign operating environment (e.g. the number and magnitude of transients is small, generally not in contact with primary water at plant operating temperatures/pressures, etc.). The robust design is manifested in that plant operation has been allowed at several plants even with a bolt/stud assumed to

be out of service. As such, significant degradation of multiple bolts/threads would be needed prior to any RCS leakage.

With the above in mind, an assessment was performed which bounded any impact of eliminating the RPV flange ligament examination which is discussed as follows:

In order to determine the impact on risk, a determination of initiating event (IE) and conditional core damage probability (CCDP) given the event is needed. A survey was conducted to determine representative values for use as a CCDP. If leakage were to occur, it would be most likely very limited and result in less leakage compared to events modeled as small LOCA in plant risk assessments (PRAs). As such, small-small LOCA (very small LOCA, VSLOCA), normal plant trip (NPT) and manual plant shutdown (MSHTDN) was used to bound the CCDP value. (Note, it is believed that manual plant shutdown is the more representative case.) These assumptions are based upon the significant operating and inspection experience as documented in the EPRI report. Thus, an upper bound value for CCDP is 7E-5 for VSLOCA, a value of 2E-06 for NPT and a lower bound value of <1E-06 for MSHTDN was used.

With respect to the initiating event, there have been no occurrences of pressure boundary leakage related to the RPV flange ligament inspection. Assuming one event in the 16,000 reactors years of operation, allows a conservative assessment of the benefit of continuing the RPV Flange ligament examinations. And while it is anticipated that with the elimination of the examinations, there will continue to be no leakage, an upper bound of 1 event per 10 years with 100 plants in the US industry is used for the case with the ligament examination requirements no longer in effect. The basis for this assumption is that if the event were to occur, per industry operating experience review requirements, other applicable plants would need to assess the applicability of the event to their operating regime/practices prior to any significant new occurrences. Additionally, due to the size of the US fleet and their associated fuel cycle (12 month, 18 month, 24 month), a large number of plants undergo refueling outages each year. As such, there is ample opportunity, irrespective of volumetric examination of the RPV flange ligament, to assess the overall condition of the RPV flange connection visually during the refueling outage. Therefore, the initiating event frequency for the base case (i.e. continue with the existing ligament examination requirements) will be conservatively assumed to be 6.3E-05 (i.e. 1/16,000) and the frequency for the proposed action (i.e. elimination of the examination requirement) will be 1.0E-03 (i.e. 1/ (10*100)).

Based on the above, the risk associated with the proposed action is provided below. As can be seen, the risk captured by the current requirements is extremely low and not commensurate with the associated burden (worker exposure, personnel safety, critical path time). As such, even with the conservative assumptions discussed above, the impact on risk from an industry wide perspective is quite low.

In addition, it is important to note all other inspection activities, including the system leakage test (examination category B-P), which is conducted each refueling outage, will continue to be performed.

Finally, as discussed in Reference 2, which includes work supported by the USNRC, without an active degradation mechanism present, it was concluded that if preservice inspection has confirmed that the inspection volume is in good condition (i.e. no flaws / indications), then subsequent inservice inspections are not providing any additional value added going forward. As stated earlier, the RPV flange ligaments have received the required pre-service examinations, but over 10,000 inservice inspections, with no relevant findings.

Table 2: Bounding Risk Captured By Current Requirements

Current		$IE_{frequency}$		CCDP		CDF
MSHTDN		6.3E-05		<1.0E-06		<6.3E-11
NPT		6.3E-05		2.0E-06		1.3E-10
VSLOCA		6.3E-05		7.0E-05		4.4E-09

Table 3: Bounding Risk Increase Due to the Proposed Action

Proposed Action		$IE_{frequency}$		CCDP		CDF
MSHTDN		1.0E-03		<1.0E-06		<1.0E-09
NPT		1.0E-03		2.0E-06		2.0E-09
VSLOCA		1.0E-03		7.0E-05		7.0E-08

Proposed Alternative and Basis for Use:

In lieu of the requirements for a volumetric ultrasonic examination, SNC is proposes that the industry report provides an acceptable technical basis for eliminating the requirement for this examination for VEGP.

This report provides the basis for the elimination of the RPV Threads in Flange examination requirement (ASME Section XI Examination Category B-G-1, Item Number B6.40). This report was developed because evidence had suggested that there have been no occurrences of service-induced degradation and there are negative impacts on worker dose, personnel safety and critical path time for these examinations. The report also included an industry survey plus an assessment of the structural integrity of the RPV and a bounding generic risk assessment.

Since there is reasonable assurance that the proposed alternative is an acceptable alternate approach to the performance of the ultrasonic examinations, SNC requests authorization to use the proposed alternative pursuant to 10 CFR 50.55a(z)(1) on the basis that the use of the alternative provides an acceptable level of quality and safety.

To protect against non-service related degradation, each Outage SNC uses a detailed procedure for the removal, care and visual inspection of the RPV studs and the threads in flange. Care is taken to not only remove the studs, but once the studs are removed to inspect the RPV threads for damage and to install RPV stud plugs to protect threads from damage. Prior to reinstallation, the studs and stud holes are cleaned and lubricated. The studs are then replaced and tensioned into the Reactor Vessel. This activity is performed each refueling outage and the procedure documents each step. These controlled maintenance activities provide further assurance that degradation is detected and mitigated prior to returning the reactor to service.

Duration of Proposed Alternative:

The alternative is requested for the current Third Inservice Inspection Interval, which began May 31, 2007 and is scheduled to end on May 30, 2017 for both Unit 1 and 2.

References:

1. EPRI Nondestructive Evaluation Report – Reactor Pressure Vessel Threads in Flange Examination Requirements. 3002007626; Dated: March 2016 Electric Power Research Institute (EPRI).

2. American Society of Mechanical Engineers, Risk-Based Inspection: Development of Guidelines, Volume 2-Part 1 and Volume 2-Part 2, Light Water Reactor (LWR) Nuclear Power Plant Components. CRTD-Vols. 20-2 and 20-4, ASME Research Task Force on Risk-Based Inspection Guidelines, Washington, D.C., 1992 and 1998.

Status:

Pending NRC approval.