

Barry S. Allen  
Vice President - Nuclear

419-321-7676  
Fax: 419-321-7582

May 20, 2011  
L-11-097

10 CFR 50.90

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

**SUBJECT:**

Davis-Besse Nuclear Power Station, Unit No. 1  
Docket No. 50-346, License No. NPF-3  
License Amendment Request to Modify the Special Visual Inspection Requirements of  
Technical Specification 5.5.8, "Steam Generator (SG) Program"

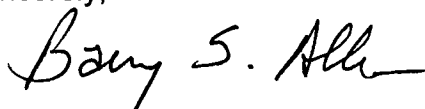
Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company requests an amendment to the Davis-Besse Nuclear Power Station, Unit No. 1 Technical Specification (TS) 5.5.8, "Steam Generator (SG) Program." The proposed change would revise TS 5.5.8.g to perform the special visual inspection based on a condition rather than a specific frequency.

An evaluation of the proposed amendment is enclosed. In order to support the refueling outage that begins on May 20, 2012, NRC staff approval is requested by May 1, 2012. An implementation period of 30 days is requested, following the effective date of the amendment.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 761-6071.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 20, 2011.

Sincerely,



Barry S. Allen

Enclosure:  
Evaluation of Proposed License Amendment

A001  
NRC

Davis-Besse Nuclear Power Station, Unit No. 1  
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cc: NRC Region III Administrator  
NRC Resident Inspector  
NRC Project Manager  
Executive Director, Ohio Emergency Management Agency,  
State of Ohio (NRC Liaison)  
Utility Radiological Safety Board

EVALUATION OF PROPOSED LICENSE AMENDMENT  
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Subject: License Amendment Request to Modify the Special Visual Inspection Requirements of Technical Specification 5.5.8, "Steam Generator (SG) Program."

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## 1.0 SUMMARY DESCRIPTION

This evaluation supports a FirstEnergy Nuclear Operating Company (FENOC) request to modify the special visual inspection frequency requirements of Davis-Besse Nuclear Power Station (DBNPS) Technical Specification (TS) 5.5.8, "Steam Generator (SG) Program." Specifically, TS 5.5.8.g requires visual inspections of the secured internal auxiliary feedwater header, header to shroud attachment welds, and external header thermal sleeves. These inspections are required to be performed during the third period of each ten-year inservice inspection interval. The proposed license amendment would replace the time-based frequency requirement with a condition-based inspection requirement. With the proposed change, if the eddy current inspections required by TS 5.5.8.d.5 identify any steam generator peripheral tube to secured internal auxiliary feedwater header gap less than 1/4 inch, then the TS 5.5.8.g inspections shall be performed on the affected steam generator. Modifying this requirement is predicated on historically acceptable visual and eddy current inspection results for both steam generators and an evaluation of changes to operational conditions. It does not involve a design modification or physical change to the plant, and it does not change methods of plant operation or maintenance of equipment important to safety.

## 2.0 DETAILED DESCRIPTION

### Internal Auxiliary Feedwater Headers

The internal auxiliary feedwater headers (AFWHs) were originally a part of the auxiliary feedwater system (AFWS). Both steam generators 1-B and 2-A were designed and fabricated with internal AFWHs. The AFWH consists of a rectangular-shaped flow channel that fully encircles the steam generator tube bundle. The AFWHs are located at and are supported from the top of the steam shroud between the 15th tube support plate and the upper tube sheet. The primary function of the AFWH was to uniformly distribute auxiliary feedwater flow inside the steam generator through 60 equally-spaced holes surrounding the steam generator tube bundle. The AFWH also functions as an extension of the upper shroud, which separates the tube bundle from the steam outlet annulus. The internal AFWHs no longer function as part of the AFWS. The AFWHs in both steam generators were secured and abandoned in-place in 1982, as discussed below. Access to the internal AFWHs for performing the TS 5.5.8.g special visual inspections is limited. For each steam generator, it requires the removal of eight external AFWH risers and their associated thermal sleeves. A remotely operated video camera or fiber optic device is then required to perform the inspections. A small portion of the AFWH is also visually accessible upon removal of the steam generator's upper secondary manway cover.

### Steam Generator Tube Inspections and Identified Damage

During a 1982 refueling outage at DBNPS, steam generator eddy current inspections identified indications in some of the steam generator peripheral tubes. The individual indications correlated with the top and bottom edges of the internal AFWH. Secondary side visual inspections identified that the internal AFWH in each steam generator was dislodged, severely deformed, and damaged some of the steam generator peripheral tubes. This damage was initially documented in a licensee event report dated May 24, 1982 [Reference 3].

Due to the damage, an expanded inspection program was initiated for both steam generators 1-B and 2-A. This program, documented in correspondence to the Nuclear Regulatory Commission (NRC) dated August 6, 1982 [Reference 4], included:

- 100 percent peripheral tube eddy current inspections
- Selected profilometry inspection of peripheral tubes with eddy current indications in the header region

The eddy current inspections revealed that a total of 24 peripheral tubes in the two steam generators had indications or denting interpreted to show contact with the internal AFWH at some point in time.

Profilometry inspections of the peripheral tubes revealed that the denting was oriented towards the internal AFWH.

Analysis of the peripheral tube eddy current inspection results revealed that the internal AFWH was dislodged and had shifted location with respect to the tube bundle.

### Damage Resolution

Due to the damage, repairs to the steam generators were required. This included plugging and stabilizing 3 of 24 steam generator peripheral tubes due to the depth of the indications, and securing and abandoning the damaged internal AFWHs. The internal AFWHs were recentered prior to fastening the header to the shroud to ensure the minimum clearance between any part of the header and an unplugged, unstabilized steam generator tube was not less than 1/8 inch. The header was actually repositioned as close to the original, nominal clearance of 9/16 inch to 2 inches. This minimum 1/8 inch clearance bounds expected flow induced vibration and thermal motions. Each AFWH was then stabilized and secured in-place with eight SA-36 carbon steel gusset plates and eight Type P1 carbon steel fillet welds. The gusset plates and attachment welds were located above each of the eight new AFWS injection penetrations. A three-dimensional analysis using the ANSYS Finite Element Code ensured the adequacy of the stabilized header and its attachment system. After completion of the repairs, the internal AFWHs continued to function as an extension of the tube bundle upper shroud, but no longer as part of the AFWS. The internal AFWHs were abandoned in-place because existing construction features of the steam generators made removal of the damaged AFWHs extremely difficult.

Due to the damage, plant modifications were required. For each steam generator, this included eight new penetrations through the steam generator's shell and steam shroud. It also included installing two new external AFWHs, each consisting of eight header risers and tapered thermal [injection] sleeves. For each steam generator, blind flanges were installed on the original auxiliary feedwater penetrations. The AFWS, after incorporating this modification, became similar to other designs already in service at other licensees.

Repairs and modifications adhered to applicable code requirements and an approved design change process. Safety-related and seismic requirements remained unchanged.

The NRC reviewed these repairs and modifications, and their potential impact on plant operation. The evaluation concluded that the repairs and modifications were adequate and that it was acceptable for DBNPS to resume normal plant operation. As part of the resolution, DBNPS also committed to inspect the external thermal sleeves, and the stabilized internal AFWH through the external AFWH injection openings. These commitments to inspect, and to apply for a technical specifications amendment, are documented within correspondence to the NRC dated August 6, 1982 [Reference 4]. The NRC's conclusions regarding the repairs, modifications, and resumption of normal plant operations are documented in a safety evaluation report issued on August 20, 1982 [Reference 5].

On September 30, 1983, the NRC issued Amendment 62 [Reference 9], which incorporated the special visual and eddy current inspection requirements into the DBNPS Technical Specifications. The visual inspections include the secured AFWH, the header to shroud attachment welds, and the external header thermal sleeves. The visual inspections are performed on each steam generator through the external AFWH injection penetrations, after disassembly of the external header risers and removal of the thermal sleeves. The visual inspections are performed during the third period of each ten-year inservice inspection interval. These visual inspections confirm that the AFWHs, the structural welds, and the external header thermal sleeves have not degraded during plant and system operation. The eddy current inspections are performed on at least 150 steam generator peripheral tubes designated as a special interest group. These eddy current inspections assess the gap between the secured AFWH and the steam generator peripheral tubes to ensure the AFWH is not moving within 1/4 inch of the inservice steam generator tubes and therefore does not pose a threat to any inservice steam generator tubes. These eddy current inspections are performed every refueling outage on both steam generators. The DBNPS application to incorporate these inspections into the technical specifications [Reference 8] is evaluated in the NRC's safety evaluation report [Reference 9].

### Oconee Unit 3: Similar Damage, Resolution, and Inspection Results

During a 1982 refueling outage at the Duke Power Company Oconee Nuclear Station, Unit 3, steam generator secondary side visual inspections identified internal AFWH damage similar to that reported at DBNPS. These inspections were performed on steam generators ONS-3A and ONS-3B following receipt of information that damage had been identified in similar steam generators manufactured by Babcock & Wilcox, at DBNPS. Inspections identified that the internal AFWH in each steam generator was dislodged, severely deformed, and damaged some of the steam generator peripheral tubes. Duke initially reported their steam generator inspection and damage findings in a preliminary notification of event document dated April 30, 1982 [Reference 1], and supplemented their findings with a reportable occurrence report dated May 14, 1982 [Reference 2]. For Oconee Units 1 and 2, the steam generators were of a different design without an internal AFWH; therefore, the secondary side visual inspections were not required.

Repairs and modifications completed at Oconee Unit 3 were similar to those completed at DBNPS. For steam generators ONS-3A and ONS-3B, these included securing the internal AFWH and abandoning it in-place, and installing an external feedwater header with multiple injection nozzles. The NRC reviewed the repairs and modifications and their potential impact on plant operation. The NRC's evaluation concluded that the

repairs and modifications were adequate, and that it was acceptable for Oconee Unit 3 to resume normal plant operation.

As part of the resolution, Duke Power committed to perform eddy current inspections on selected special interest steam generator peripheral tubes. Duke Power also committed to perform visual inspections of the internal header, the attachment welds, and the external header thermal sleeves. Details of these two commitments to inspect are documented within correspondence dated September 10, 1982 [Reference 6]. The NRC's conclusions regarding the resumption of plant operations are documented in a safety evaluation report issued to Duke Power on September 29, 1982 [Reference 7]. Duke Power did not apply for an amendment to incorporate these eddy current and visual inspection requirements into their plant technical specifications.

For Oconee Unit 3, the visual inspection results for steam generators ONS-3A and ONS-3B were similar to the visual results for DBNPS steam generators 1-B and 2-A. Duke Power completed the requisite visual inspections with no degradation noted. The last steam generator visual inspections were completed during a 1992 outage.

Based on acceptable eddy current and visual inspection results since the repairs and modifications were implemented in 1982, along with the continuance of peripheral tube eddy current inspections until steam generator replacement, Duke Power decided to discontinue performance of the visual inspections. This decision affected the last remaining visual inspections prior to the replacement of both steam generators. Duke Power subsequently closed their inspection commitments. For Oconee Unit 3, the steam generators have since been replaced with a design that does not include an internal AFWH.

#### Qualification of the Eddy Current Technique

The TS 5.5.8.d.5 eddy current inspections are performed on peripheral tubes in both steam generators 1-B and 2-A. These inspections are specifically targeted to detect degradation to in-service tubing due to interaction with the secured internal AFWH. Additionally, these peripheral tubes are examined to a qualified technique for obtaining proximity measurements to the internal AFWH. The technique uses standard bobbin probe data to detect carbon steel [internal AFWH] material in close proximity to the outer surface of the steam generator peripheral tubes and to assess the associated gaps. This technique is qualified for the detection of gaps as large as 0.250 inch and is applicable for both parent and sleeved tubes.

#### Stabilized and Plugged Steam Generator Peripheral Tubes

As stated above, a total of three steam generator peripheral tubes were stabilized and plugged as a result of the 1982 inspection findings.

Since 1982, peripheral tubes that required plugging due to operational conditions were also stabilized, except for two. For steam generator 1-B, one peripheral tube was sleeved and subsequently plugged. When sleeved, a new pressure boundary component is inserted into the tube allowing it to remain in service. Sleeved tubes prohibit installation of a stabilizer; however, the sleeve itself functions as a stabilizer. This tube was eventually plugged for degradation elsewhere in the tube. For steam generator 2-A, one peripheral tube was plugged, but not stabilized. This tube has a welded plug. In lieu of removing the plug to stabilize this tube, it was surrounded and protected by other tubes that were stabilized and plugged.

### Alternate Inspection Capabilities

During development of this request, an alternate internal AFWH inspection option was evaluated. As stated above, a small portion of the AFWH is directly accessible upon removal of the steam generator's upper secondary manway cover. However, the ability to obtain adequate inspections of the AFWH and the attachment welds on the opposite side of the steam generator (approximately 15 feet circumferentially away) from this location was determined to have no guarantee of success. This alternative was also estimated to incur more cumulative occupational radiation exposure [dose] than performing the current TS 5.5.8.g inspections. Additionally, inspections from this location do not meet the current requirements of TS 5.5.8.g in that the external AFWH thermal sleeves are not removed and fully inspected. FENOC concluded that this alternative would be difficult to perform, may not provide the requisite inspection coverage, and would require a change to the current inspection requirements of TS 5.5.8.g. Therefore, this alternative was not pursued.

### One-Time Amendment or Permanent Amendment

During development of this request, FENOC evaluated the merits of a permanent amendment as compared to a one-time amendment for modifying the frequency requirements of TS 5.5.8.g. FENOC concluded that the basis for either request is the same for both. Therefore, FENOC has elected to pursue a permanent change to TS 5.5.8.g, which will remain in place until the scheduled replacement of the steam generators in 2014.

### Cost and Dose for Performing TS 5.5.8.g Visual Inspections

The cost for performing the TS 5.5.8.g visual inspections of steam generator 2-A is estimated to be approximately \$300,000. The cumulative occupational radiation exposure [dose] for performing the TS 5.5.8.g visual inspections of steam generator 2-A during the spring 2012 refueling outage is estimated to be approximately 1 man-Rem.

### Technical Specification Change

FENOC proposes to modify the inspection frequency requirements of TS 5.5.8.g. Currently, TS 5.5.8.g requires the special visual inspections to be performed during the third period of each ten-year inservice inspection interval. With the proposed change, if the inspections required by TS 5.5.8.d.5 identify any steam generator peripheral tube to secured internal AFWH gap less than 1/4 inch, then the TS 5.5.8.g inspections shall be performed on the affected steam generator.

The TS 5.5.8.d.5 eddy current inspections are performed on both steam generator 1-B and 2-A during every refueling outage. Tubes selected for inspection must represent the entire circumference of each steam generator and shall total at least 150 peripheral tubes per steam generator. The inspections assess the gap between peripheral tubes and the secured internal AFWH, and the results are compared against the 1983 baseline criteria.



Incorporation of the TS 5.5.8.d.5 eddy current and TS 5.5.8.g special visual inspection requirements into the technical specifications were the direct result of a 1982 DBNPS operational event. The TS 5.5.8.g special visual inspections are not a requirement of the American Society of Mechanical Engineers (ASME) Code. Historically, these visual inspections have been performed as VT-3 examinations.

Attachment 1 provides TS 5.5.8, including TS 5.5.8.d.5 and a marked-up version of TS 5.5.8.g; Attachment 2 provides a retyped version of TS 5.5.8.g.

### 3.0 TECHNICAL EVALUATION

As required by TS 5.5.8.g, special visual inspections of the internal AFWH, the header to shroud attachment welds, and the external header thermal sleeves are performed on both steam generators. These visual inspections are required to be performed during the third period of each ten-year inservice inspection interval. The visual inspections verify that degradation of the internal AFWH and external header thermal sleeves have not occurred during plant and system operation. The visual inspections have been performed, as required, since the discovery of the internal AFWH damage in 1982. FENOC proposes to change this requirement. With the proposed change, if the inspections required by TS 5.5.8.d.5 identify any steam generator peripheral tube to secured internal AFWH gap less than 1/4 inch, then the TS 5.5.8.g inspections shall be performed on the affected steam generator. The 1/4 inch criteria was established to provide sufficient margin to the 1/8 inch minimum clearance required for flow induced vibration and thermal movement concerns.

For steam generator 1-B, four small linear indications on external header thermal sleeve no. 3 were identified during the visual inspections. The indications, identified during the 1984 refueling outage, appeared to be the result of visual inspection activities (disassembly and reassembly), not plant or system operational conditions. Following rework of the thermal sleeve and its reinstallation, no other recordable indications have been identified since 1984. For steam generator 1-B, the last visual inspection was completed during the 2010 refueling outage, and no other TS 5.5.8.g special visual inspections remain before the steam generator's scheduled replacement in 2014.

For steam generator 2-A, no recordable indications have been identified during the special visual inspections. For steam generator 2-A, the last visual inspection was performed during the 1998 refueling outage, and only one TS 5.5.8.g special visual inspection remains before the end of the current inservice inspection interval (scheduled to expire September 20, 2012).

As required by TS 5.5.8.d.5, eddy current inspections of peripheral tubes in the vicinity of the secured internal AFWHs are performed on both steam generator 1-B and 2-A during every refueling outage. Tubes selected for inspection represent the entire circumference of each steam generator and total at least 150 peripheral tubes. This special interest inspection assesses the existing gap between the secured internal AFWH and peripheral tubes to ensure the AFWH is not moving and is not a threat to any inservice steam generator tubes. This TS 5.5.8.d.5 eddy current inspection has been performed on both steam generators during every refueling outage since the discovery of damage in 1982. Significant changes in gaps have not been detected since establishment of the gap baseline criteria in 1983. This includes the most recent TS 5.5.8.d.5 eddy current inspections of steam generator 1-B and 2-A completed during the 2010 refueling outage. Typically, all in-service peripheral tubes in each steam

generator have been selected for inspection. These inspections will continue in accordance with plant technical specifications.

Attachment 3 summarizes the historical results of the TS 5.5.8.g visual and TS 5.5.8.d.5 eddy current inspections for both steam generators. As the tabular results depict, acceptable visual inspections have been obtained since 1984. For the eddy current inspections, no significant changes in gap conditions have been identified since 1983.

In support of the 2007 turbine header pressure modification, steam generators 1-B and 2-A were evaluated for the expected changes in operational conditions. The evaluations concluded that the increase in header pressure from 870 pounds per square inch gauge (psig) to 880 psig resulted in a negligible affect on flow induced vibration and would not significantly impact the steam generators. In support of the 2008 reactor power uprate, steam generators 1-B and 2-A were again evaluated for the expected changes in operational conditions. The evaluations concluded that the approximate 1.6 percent reactor power increase resulted in a negligible effect on flow induced vibration and would not significantly impact the steam generators. The acceptable 2010 special visual inspections of steam generator 1-B, and eddy current inspections of steam generators 1-B and 2-A, validate both the header pressure change and power uprate evaluations. No other design modifications with the potential to affect the steam generators, the secured and abandoned internal AFWHs, or the external header thermal sleeves, were identified.

Since 1998, several operational events with a potential to damage the steam generators, the steam generator tubes, the secured and abandoned internal AFWHs, or the external header thermal sleeves, have occurred. These events occurred either in steam generator 1-B or both steam generators 1-B and 2-A. Follow-up inspections after the individual events did not identify any damage. The acceptable 2010 special visual inspections of steam generator 1-B, and eddy current inspections of steam generators 1-B and 2-A, validate the results of the individual post-event inspections.

The remaining planned service life for both steam generators is short. FENOC is scheduled to replace the current once-through steam generators in 2014. The upgraded steam generators will not have an internal AFWH but will have external header thermal sleeves. These thermal sleeves will not require the visual inspections prescribed in TS 5.5.8.g as the degradation mechanism of the current steam generator [damaged and secured AFWHs, gusset plates and welds] will not be present. For the current steam generators, only one TS 5.5.8.g inspection of steam generator 2-A remains before its scheduled replacement. That inspection is currently scheduled during the spring 2012 refueling outage.

Modifying the special visual inspection frequency requirement does not involve a design modification or physical change to the plant, nor does it change methods of plant operation or maintenance of equipment important to safety. Past visual inspections have verified that internal AFWH and thermal sleeve degradation has not occurred. This acceptable inspection trend indicates degradation has not occurred, would not be expected to be identified in the remaining required inspections, and is not expected to occur during the remaining service life of the steam generators. Therefore, modification of the inspection requirements of TS 5.5.8.g is warranted.

#### 4.0 REGULATORY EVALUATION

The proposed amendment would modify the Davis-Besse Nuclear Power Station frequency requirement to perform visual inspections of the internal AFWH, the header to shroud attachment welds, and the external header thermal sleeves, for both steam generators 1-B and 2-A. Currently, TS 5.5.8.g requires the special visual inspections to be performed during the third period of each ten-year inservice inspection interval. With the proposed change, if the inspections required by TS 5.5.8.d.5 identify any peripheral tube to secured internal AFWH gap less than 1/4 inch, then the TS 5.5.8.g inspections shall be performed on the affected steam generator. Modifying the inspection frequency requirement is predicated on acceptable visual and eddy current inspection results with no degradation identified, an evaluation of changes to operational conditions, and an inspection trend that indicates that degradation is not expected to occur during the remaining planned service life of the steam generators. The proposed amendment does not involve a design modification or physical change to the plant, and does not change methods of plant operation or maintenance of equipment important to safety.

##### 4.1 No Significant Hazards Consideration Analysis

FirstEnergy Nuclear Operating Company (FENOC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

This amendment request modifies the frequency requirements for performing visual inspections of the secured internal auxiliary feedwater header (AFWH), the header to shroud attachment welds, and the external header thermal sleeves. Previous AFWH repairs adhered to applicable code requirements, and steam generator modifications to install an external AFWH were implemented using an approved design change process. The steam generator's safety-related and seismic design requirements did not change. The auxiliary feedwater system's safety-related, seismic and accident mitigation requirements did not change.

Historically acceptable TS 5.5.8.g visual inspection results for both steam generators indicate no degradation of the internal AFWH and its attachment welds, or of the external header thermal sleeves. Historically acceptable TS 5.5.8.d.5 eddy current inspections indicate no degradation of the steam generator peripheral tubes due to the internal AFWH. The acceptable eddy current inspections were last completed on both steam generators during the 2010 refueling outage, and will continue to be performed until the steam generators are replaced. For steam generator 1-B, the last TS 5.5.8.g special visual inspection was completed during the 2010 refueling outage, and no other required TS 5.5.8.g special visual inspections remain before the steam generator's scheduled replacement in 2014. For steam generator 2-A, the last TS 5.5.8.g special visual inspection was performed during the 1998 refueling outage, and only one TS 5.5.8.g special visual inspection remains before the end of the current inservice inspection interval (September 20, 2012).

The visual inspection trend indicates that degradation of the secured internal AFWH and its attachment welds is not expected to be identified, should the required TS 5.5.8.g inspection of steam generator 2-A be performed in 2012, and that degradation is not expected to occur during the remaining planned service life of both steam generators before their scheduled replacements in 2014.

The eddy current inspection trend indicates that the steam generator tubes, a part of the reactor coolant system, remain unaffected by the internal AFWHs. Significant changes in the gap between the steam generator peripheral tubes and the secured AFWHs is not expected to be identified when the next TS 5.5.8.d.5 eddy current inspections of steam generators 1-B and 2-A are performed in 2012, and that significant gap degradation is not expected to occur during the remaining planned service life of both steam generators before their scheduled replacements in 2014.

Should significant damage to a steam generator peripheral tube occur due to direct contact with the internal AFWH, the postulated occurrence would be bounded by the existing Updated Safety Analysis Report (USAR) Chapter 15, Section 15.4.2, for the steam generator tube rupture accident analysis.

Modifying the TS 5.5.8.g inspection frequency requirement will not significantly increase the probability of an accident previously evaluated because historically acceptable inspection results and trend indicate that degradation of the secured internal AFWHs is not expected to occur during the remaining service life of both steam generators.

Modifying the TS 5.5.8.g inspection frequency requirement will not cause a change to any of the dose analyses associated with the USAR Chapter 15 accidents because accident mitigation functions and requirements remain unchanged.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

This amendment request modifies the frequency requirements for performing visual inspections of the secured internal AFWH, the header to shroud attachment welds, and the external header thermal sleeves. This request does not change the design function of the reactor coolant system, the steam generators, the AFWS, or the way the systems and plant are operated and maintained.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

This amendment request modifies the frequency requirements for performing visual inspections of the secured internal AFWH, the header to shroud attachment welds, and the external header thermal sleeves. Historically acceptable TS 5.5.8.g visual inspection results for both steam generators indicate no degradation of the internal AFWH and its attachment welds, or of the external header thermal sleeves. Historically

acceptable TS 5.5.8.d.5 eddy current inspections indicate no degradation of the steam generator [special interest group] peripheral tubes due to the internal AFWH.

The visual and eddy current inspection trends indicate that degradation of the AFWH, or significant changes in the gap between the steam generator peripheral tubes and the secured AFWHs, is not expected to occur during the remaining service life of the steam generators before their scheduled replacements in 2014.

This request does not involve or affect the fuel cladding or the containment. The steam generator tubes, a part of the reactor coolant system, remain unaffected based on the historically acceptable TS 5.5.8.d.5 eddy current inspection results and trend. This request does not involve a physical change to the plant, methods of plant operation, or maintenance of equipment important to safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FENOC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

#### 4.2 Applicable Regulatory Requirements/Criteria

The proposed amendment has been evaluated against the following NUREG-0800 Standard Review Plan sections to determine whether applicable regulations and requirements would continue to be met.

- Section 5.4, Reactor Coolant System Component and Subsystem Design
- Section 5.4.2.1, Steam Generator Materials
- Section 5.4.2.2, Steam Generator Program
- Section 10.3, Main Steam Supply System
- Section 10.4.9, Auxiliary Feedwater System (PWR)

FENOC has determined that the proposed amendment does not require any exemptions or relief from regulatory requirements, and does not affect conformance with any General Design Criterion differently than described in the NUREG sections or in DBNPS's USAR.

#### 4.3 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

## 5.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

## 6.0 REFERENCES

1. Duke Power Company, Oconee Nuclear Station, Unit 3, Preliminary Notification of Event or Unusual Occurrence PN-II-82-47A, April 30, 1982.
2. Duke Power Company letter to U.S. Nuclear Regulatory Commission, Subject: Oconee Nuclear Station, Unit 3, Reportable Occurrence Report RO-287/82-06, May 14, 1982.
3. Toledo Edison Company, Davis-Besse Nuclear Power Station Unit One, Licensee Event Report No. 82-19, May 24, 1982.
4. Toledo Edison Company letter to U.S. Nuclear Regulatory Commission, Subject: Davis-Besse Nuclear Power Station Unit No. 1, August 6, 1982.
5. U.S. Nuclear Regulatory Commission letter to Toledo Edison Company, Subject: Once-Through Steam Generator Repair and Auxiliary Feedwater System Modification – Safety Evaluation Report, August 20, 1982.
6. Duke Power Company letter to U.S. Nuclear Regulatory Commission, Subject: Oconee Nuclear Station, Unit 3, Report on Oconee Unit 3 Auxiliary Feedwater Header Repair, September 10, 1982.
7. U.S. Nuclear Regulatory Commission letter to Duke Power Company, Subject: Once-Through Steam Generator (SG) Repair and Emergency Feedwater (EFW) System Modification – Safety Evaluation Report, September 29, 1982.
8. Toledo Edison Company letter to U.S. Nuclear Regulatory Commission, Subject: Davis-Besse Nuclear Power Station Unit No. 1, License Amendment Request, January 12, 1983.
9. U.S. Nuclear Regulatory Commission letter to Toledo Edison Company, Subject: Amendment No. 62 to Facility Operating License No. NPF-3; Steam Generator Inservice Inspection, September 30, 1983.

Attachment 1

**PROPOSED TECHNICAL SPECIFICATION CHANGES**

**(MARK-UP)**

(Four Pages Follow)

## 5.5 Programs and Manuals

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### 5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

- a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met.
- b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.
  1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.
  2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG except during a steam generator tube rupture.



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5.5.8 Steam Generator (SG) Program (continued)

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."
- c. Provisions for SG tube repair criteria:
1. Tubes found by inservice inspection to contain flaws, in a region of the tube that contains no repair, with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired;
  2. Sleeves found by inservice inspection to contain flaws, in a region of the sleeve that contains no sleeve joint, with a depth equal to or exceeding 40% of the nominal sleeve wall thickness shall be plugged;
  3. Tubes with a flaw, in either parent tube or the sleeve, within a sleeve to tube joint shall be plugged; and
  4. Tubes with a flaw in a repair roll shall be plugged.
- d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. For tubes that have undergone repair rolling, the tube and tube roll, outboard of the new roll area in the tube sheet, can be excluded from inspections because it is no longer part of the pressure boundary once the repair roll is installed. For tubes that have undergone sleeving repairs, the segment of the parent tube between the upper-most sleeve roll and the top of the middle sleeve roll can be excluded from inspection because it is no longer part of the pressure boundary once the sleeve is installed. In addition to meeting the requirements of d.1 through d.5 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.
1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

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5.5.8 Steam Generator (SG) Program (continued)

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.
  3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.
  4. During each periodic SG tube inspection, inspect 100% of the tubes that have been repaired by the repair roll process. This special inspection shall be limited to the repair roll joint and the roll transitions of the roll repair.
  5. Inspect peripheral tubes in the vicinity of the secured internal auxiliary feedwater header between the upper tube sheet and the 15th tube support plate during each periodic SG tube inspection. The tubes selected for inspection shall represent the entire circumference of the steam generator and shall total at least 150 peripheral tubes.
- e. Provisions for monitoring operational primary to secondary LEAKAGE.
- f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.
1. Sleeving in accordance with Topical Report BAW-2120P.
  2. Repair rolling in accordance with Topical Report BAW-2303P, Revision 4. The new roll area must be free of flaws in order for the repair to be considered acceptable.

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5.5.8 Steam Generator (SG) Program (continued)

- g. ~~Special visual inspections: Visual inspections of the secured internal auxiliary feedwater header, header to shroud attachment welds, and the external header thermal sleeves shall be performed on each SG through the auxiliary feedwater injection penetrations. These inspections shall be performed during the third period of each 10-year Inservice Inspection Interval (ISI).~~

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to

**INSERT:**

Special visual inspections: If the inspections required by TS 5.5.8.d.5 identify any peripheral tube to secured internal auxiliary feedwater header gaps less than 1/4 inch, visual inspections of the secured internal auxiliary feedwater header, header to shroud attachment welds, and external header thermal sleeves shall be performed on the affected SG through the auxiliary feedwater injection penetrations.

variables and control  
values of the critical

- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of safety related filter ventilation systems in accordance with Regulatory Guide 1.52, Revision 2, ANSI/ASME N510-1980, and ASTM D 3803-1989.

- a. Demonstrate for each of the safety related systems that an in-place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-1980 at the system flowrate specified below.

<u>Safety Related Ventilation System</u>	<u>Flowrate (cfm)</u>
Station Emergency Ventilation System (EVS)	≥ 7200 and ≤ 8800
Control Room Emergency Ventilation System (CREVS)	≥ 2970 and ≤ 3630

Attachment 2

**PROPOSED TECHNICAL SPECIFICATION CHANGES**

**(RETYPED)**

(One Page Follows)

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5.5.8 Steam Generator (SG) Program (continued)

- g. Special visual inspections: If the inspections required by TS 5.5.8.d.5 identify any peripheral tube to secured internal auxiliary feedwater header gaps less than 1/4 inch, visual inspections of the secured internal auxiliary feedwater header, header to shroud attachment welds, and external header thermal sleeves shall be performed on the affected SG through the auxiliary feedwater injection penetrations.

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

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<u>Safety Related Ventilation System</u>	<u>Flowrate (cfm)</u>
Station Emergency Ventilation System (EVS)	≥ 7200 and ≤ 8800
Control Room Emergency Ventilation System (CREVS)	≥ 2970 and ≤ 3630

**SUMMARY TABLE OF STEAM GENERATOR INSPECTIONS**

Inservice Inspection Interval	Inspection Year	Steam Generator		Steam Generator	
		1-B	2-A	1-B	2-A
		TS 5.5.8.g Special Visual Inspection		Eddy Current Inspection	
31-Jul-1978  1st 10-year ISI Interval	1980				
	1982	Steam generator tube and auxiliary feedwater header damage identified and repaired			
	1983	NRC/NRC/NRC	NRC/NRC/1 UAI	AFWH Baseline established	
	1984	NRC/NRC/1 UAI+1 RW	NRC/NRC/2 UAI	No significant change	
	1986	N/A	N/A	No significant change	
	1988	N/A	N/A	No significant change	
20-Sep-1990	1990	NRC/NRC/NRC	NRC/NRC/NRC	No significant change	
21-Sep-1990  2nd 10-year ISI Interval	1991	N/A	N/A	No significant change	
	1993	N/A	N/A	No significant change	
	1994	N/A	N/A	No significant change	
	1996	N/A	N/A	No significant change	
	1998	NRC/NRC/NRC*	NRC/NRC/NRC	No significant change	
20-Sep-2000	2000	NRC/NRC/NRC**	N/A	No significant change	
21-Sep-2000  3rd 10-year ISI Interval	2002	N/A	N/A	No significant change	
	2005	N/A	N/A	No significant change	
	2006	N/A	N/A	No significant change	
	2007	N/A	N/A	No significant change	
	2010	NRC/NRC/NRC	N/A	No significant change	
20-Sep-2012					

**TABLE ACRONYMS:**

AFW = Auxiliary Feedwater  
 AFWH = Auxiliary Feedwater Header  
 N/A = Inspection Not Performed  
 NRC = No Recordable Condition  
 RW = Rework Disposition  
 SG = Steam Generator  
 UAI = Use-As-Is Disposition

**NOTES:**

Technical Specification (TS) 5.5.8, "Steam Generator (SG) Program"  
 TS 5.5.8.g inspection results are reported as "internal AFWH / AFWH-to-shroud welds / AFW thermal sleeves."  
 Eddy current inspections during the 1983 refueling outage established the AFWH baseline conditions; subsequent inspections have not detected any further movement or degradation of the internal AFWH.

**INSPECTION DISPOSITION CLARIFICATIONS:**

1983, SG 2-A	AFWH sleeve #1 stuck in penetration, dispositioned UAI based upon other sleeve NRCs
1984, SG 1-B	AFWH sleeve #4 stuck in penetration, dispositioned UAI based upon other sleeve NRCs
	AFWH sleeve #3 exhibited 4 small linear indications, RW surface, sleeve reinstalled
1984, SG 2-A	AFWH sleeve #1 stuck in penetration, dispositioned UAI based upon other sleeve NRCs
	AFWH sleeve #8 stuck in penetration, dispositioned UAI based upon other sleeve NRCs
1998, SG 1-B	*AFWH sleeve #4 stuck in penetration, to be inspected in 2000
2000, SG 1-B	**AFWH sleeve #4, rescheduled inspection from 1998