APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NUMBER NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

Attached are the requested changes to the Davis-Besse Nuclear Power Station, Unit Number 1, Facility Operating License Number NPF-3. Also included is the Safety Assessment and Significant Hazards Consideration.

The proposed changes (submitted under cover letter Serial Number 2512) concern:

Appendix A, Technical Specification Sections (TS) 3/4.4.5, Reactor Coolant System - Steam Generators, TS 3/4.4.6.2, Reactor Coolant System - Operational Leakage, and their associated Bases.

By: _____
John K. Wood, Vice President - Nuclear

Sworn to and subscribed before me this 26th day of February, 1998.

_____
Notary Public, State of Ohio

LAURA A. JENNISON
Notary Public, State of Ohio
My Commission Expires 8-15-2001

The following information is provided to support issuance of the requested changes to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1, Facility Operating License Number NPF-3, Appendix A, Technical Specifications. The changes involve Technical Specification (TS) 3/4.4.5, Reactor Coolant System - Steam Generators, TS 3/4.4.6.2, Reactor Coolant System - Operational Leakage, and their associated Bases.

A. Time Required to Implement: This change is to be implemented within 90 days after the NRC issuance of the License Amendment.

B. Reason for Change (License Amendment Request Number 97-0016):

This application proposes to revise the TS Surveillance Requirements (SR) for steam generator tube sample selection and inspection to include the recently qualified repair roll method of steam generator tube repair. The associated SRs that support steam generator tube sample selection and inspection are proposed for change where necessary to incorporate the proposed repair roll repair method. The reduced leakage limit for primary to secondary steam generator leakage is proposed for the Operational Leakage Limiting Condition for Operation (LCO). A new surveillance method to enable accurate detection of the relatively low allowed leakage is proposed for the associated SR.

C. Safety Assessment and Significant Hazards Consideration: See Attachment 1.

D. Proprietary Framatome Technologies Incorporated Topical Report BAW-2303P, Revision 3, "OTSG Repair Roll Qualification Report," dated October 1997: See Attachment 2.

Docket Number 50-346
License Number NPF-3
Serial Number 2512
Attachment 1

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR LICENSE AMENDMENT REQUEST NO. 97-0016

(34 Pages Follow)

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR LICENSE AMENDMENT REQUEST NUMBER 97-0016

TITLE:

Proposed Modification to the Davis-Besse Nuclear Power Station (DBNPS) Operating License NPF-3, Appendix A Technical Specifications (TS) to Incorporate a New Repair Roll Process for Steam Generator Tubes With Defects Within the Upper Tubesheet

DESCRIPTION:

The DBNPS has two Once-Through Steam Generators (OTSGs) that were manufactured by Babcock and Wilcox. The OTSG tubes were fabricated from Inconel Alloy 600 material and were restrained by roll expansion joints in the upper and lower tubesheets. The original tube-to-tubesheet rolls were expanded by a hard roll process and are about 1-2 inches in axial length extended into the tubesheet. The tubesheet is about 24 inches thick and a tube seal weld is provided at the primary face of the tubesheet to prevent leakage from the primary to secondary systems.

This License Amendment Request (LAR) proposes the revision of the DBNPS Technical Specifications to incorporate the steam generator tube-to-tubesheet roll expansion (repair roll) as a repair method for the OTSGs. Tube repair roll is a repair process for OTSG degraded tubes where the degradation is located in the tube within the upper tubesheet. The process creates a new pressure boundary at the repair roll joint by roll expanding a new mechanical tube-to-tubesheet joint below the region of tube defects in the upper tubesheet. The new pressure boundary joint is installed between the degraded area of the tube and the secondary face of the tubesheet, removing the degraded area from pressure boundary service (see attached figure). The repair roll process is applied only once per affected tube. This type of repair has been qualified previously by Framatome Technologies Incorporated for Westinghouse Model 27 steam generators at Connecticut Yankee and Westinghouse Model 51 steam generators at Point Beach, DC Cook, and Indian Point. A similar LAR for the upper tubesheet was approved by the NRC for Duke Power's Oconee Nuclear Station Units 1, 2 and 3 on November 21, 1997, as License Amendments Numbers 227, 227, and 224 (TAC Numbers M99779, M99780, and M99781).

Currently, the DBNPS TS Surveillance Requirements (SR) limit the repair method for a tube that has degraded to the point of requiring repair (i.e., the repair limit) to either removal from service by plugging or repair by sleeving. This LAR requests that tube repair roll, as described in proprietary Framatome Technologies Incorporated Topical Report BAW-2303P, Revision 3, "OTSG Repair Roll Qualification Report," dated October 1997, be included as a repair option for tube defects in the upper tubesheet. Furthermore, this LAR requests that the pressure boundary joint be

defined as the tube-to-tubesheet expansion joint that is closest to the secondary face of the tubesheet. Additionally, this LAR proposes several associated administrative changes.

Each of the following proposed revisions are shown on the attached marked-up TS pages:

Revise SR 4.4.5.2.a, to remove the parenthetical phrase "(subsequent to the baseline inspection)." The baseline inspection has already been performed and continued reference to it is superfluous. This is an administrative change.

Revise SR 4.4.5.2.a.1 to add "repair roll or" as a type of repaired tube that is not required to be included in the tubes selected for the first sample of a steam generator inservice inspection. Also add "(Tubes repaired by sleeving or repair roll remain available for random selection)." This change excludes the required inspection of the repair rolled tube similar to how SR 4.4.5.2.a.1 presently excludes the required inspection of a tube repaired by plugging or sleeving. However, the tube with a repair roll will still remain available for selection, on a random basis, for inspection under SR 4.4.5.2. The required inspection of all repair roll repairs is addressed later in this LAR under a proposed new SR 4.4.5.9.

Revise SR 4.4.5.3.a to remove the following portion of the paragraph: "The baseline inspection shall be performed to coincide with the first scheduled refueling outage but no later than April 30, 1980. Subsequent...." The baseline inspection has already been performed. The existing wording is superfluous. This is an administrative change.

Revise SR 4.4.5.3.c.1 to add the following paragraph:

> If the leak is determined to be from a repair roll joint, rather than selecting a random sample, inspect 100% of the repair roll joints in the affected steam generator. If the results of this inspection fall into the C-3 category perform additional inspections in the unaffected steam generator.

This revision provides guidance for escalating the inspection scope when the inspection is required by SR 4.4.5.3.c.1 and repair roll joints are identified as the cause of the primary-to-secondary leak.

Revise SR 4.4.5.4.a.4, SR 4.4.5.4.a.6, and 4.4.5.4.a.7 to include the use of "Repair Roll" as a repair method. As described above, repair roll is being proposed by this LAR to provide a repair method that is an alternative to tube plugging. Repair roll can be used for repair of tube defects in the upper tubesheet while allowing the steam generator tube to remain in service.

Add the following paragraph to SR 4.4.5.4.a.7:

> The repair roll process will only be used to repair tubes with defects in the upper tubesheet area. The repair roll process will be performed only once per steam generator tube using a 1 inch reroll length. The new roll area must be free of degradation in order for the repair to be considered acceptable. The repair roll process used is described in the Topical Report BAW-2303P, Revision 3.

This change is proposed to limit the repair roll process to the upper tubesheet, and to limit repair roll tube repairs to one repair per steam generator tube. Also, the prescribed roll length is 1 inch as described in the Topical Report. The new roll area must be determined to be free of degradation in order for the repair to be considered acceptable. This requirement establishes the acceptance criteria for repairs made using the repair roll process. Additionally, this change identifies Topical Report BAW-2303P, Revision 3 as the topical report that describes the repair roll process to be used.

Add new page 3/4 4-9a. This new page is necessitated by the additional text added to page 3/4 4-9 by the revision of SR 4.4.5.4.a.7 and SR 4.4.5.4.a.9.

Revise 4.4.5.4.a.9 to add the following sentence:

> The previously existing tube and tube roll, above the new roll area in the upper tubesheet, can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

This revision is requested to specify the location of the pressure boundary once the repair roll is installed.

Revise SR 4.4.5.4.b to include "Repair Roll" as a repair method. Repair roll is being proposed by this LAR to provide a repair method that is an alternative to tube plugging. Repair roll can be used for repair of tube defects in the upper tubesheet while allowing the steam generator tube to remain in service.

Revise SR 4.4.5.5.b.3 to include "Repair Roll" as a repair method, similar to plugging or sleeving, to be reported to the NRC.

Add a new SR 4.4.5.9 to state the following:

> When steam generator tube inspection is performed as per Section 4.4.5.2, an additional but totally separate inspection shall be performed on special interest tubes that have been repaired by the repair roll process. This inspection shall be performed on 100% of the tubes that have been repaired by the repair roll process. The inspection shall be limited to the repair roll joint and the roll transitions of the repair roll. Defective or degraded tubes

found in the repair roll region as a result of the inspection need not be included in determining the Inspection Results Category for the general steam generator inspection.

This new SR is proposed to ensure that all newly created repair rolls installed in the steam generators receive an inspection each time an inservice inspection is performed pursuant to SR 4.4.5.2. This will provide a comprehensive monitoring of potential degradation in the repair roll regions of the repaired tubes. Any indications found in the new rolls need not be included in determining the Inspection Results Category because all repair rolls will be inspected with the results reported to the NRC.

Revise Table 4.4-2 "Steam Generator Tube Inspection," as shown, to include "repair rolling" as a repair method.

Revise Limiting Condition for Operation (LCO) 3.4.6.2.c "Operational Leakage" to reduce the present limit of 1440 gallons per day (GPD) total primary to secondary leakage through the steam generators to 150 GPD primary to secondary leakage through the tubes of any one steam generator. This change provides additional conservatism in the operation of the DBNPS and does not have an adverse effect on safety.

Add SR 4.4.6.2.1.e to state the following:
[Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:]

e. An e[illegible]tion of secondary water radiochemistry for determination of pri[illegible]o secondary leakage through the steam generators at least once pe[illegible]ours during steady state operations.

This new SR is included to reflect a more accurate method for detecting steam generator primary to secondary leakage in accordance with the more restrictive LCO limit.

Revise the Bases Section 3/4.4.5, "Steam Generators," as follows:

Include references to the repair roll method of repair and to a leakage limitation of 150 GPD through any one steam generator, instead of 1 GPM (1440 GPD) total leakage through the steam generators.

This revision is requested to provide reference, in the Bases, to the repair roll method of repair and to a steam generator primary to secondary leakage limitation of 150 GPD through any one steam generator.

Also add the following text:

> An additional repair method for degraded steam generator tubes consists of rerolling the tubes in the upper tubesheet to create a new roll area and pressure boundary for the tube. The repair roll process will ensure that the area of degradation will not serve as a pressure boundary, thus permitting the tube to remain in service. The degraded area of the tube can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed in the upper tubesheet.
>
> All tubes which have been repaired using the repair roll process will have the new roll area inspected during the inservice inspection. Defective or degraded tube indications found in the new roll area as a result of the inspection of the repair roll and any indications found in the originally rolled region of the rerolled tube need not be included in determining the Inspection Results Category for the general steam generator inspection.
>
> The repair roll process will be performed only once per steam generator tube using a 1 inch reroll length as described in the Topical Report BAW-2303P, Revision 3. Thus, multiple applications of the rerolling process to any individual tube is not acceptable. The new roll area must be free of degradation in order for the repair to be considered acceptable. After the new roll area is initially deemed acceptable, future degradation in the new roll area will be analyzed to determine if the tube is defective and needs to be removed from service. The rerolling process is described in the Topical Report BAW-2303P, Revision 3.

This Bases revision is proposed in order to provide a description of the repair roll process.

Add new page B 3/4 4-3a. This new page is necessitated by the additional text added to page B 3/4 4-3 above.

Revise Bases Section 3/4.4.6.2, "Operational Leakage," to include a reference to the new leakage limitation of 150 GPD through any one steam generator instead of 1 GPM (1440 GPD) total primary to secondary leakage limit. Also, remove the phrase "consistent with the assumptions," used in the third paragraph to differentiate that although the leakage limit is 150 GPD primary to secondary leakage through the tubes of any one steam generator, the accident analysis assumed 1 GPM (1440 GPD). The revised sentence will read "A 1 GPM total primary to secondary leakage limit is used in the analysis of these accidents." This is an administrative change.

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED:

The following systems and components are affected by this LAR: the Reactor Coolant System, Steam Generators, and the Steam Generator tubes inside of the tubesheet.

The following activities are affected: 1) The steam generator tube repair process wherein a new roll expansion joint is created to serve as a pressure boundary when the existing roll expansion joint is determined to be degraded or a defect is determined to exist in the tube within the upper tube sheet, 2) Primary to secondary leakage surveillance activities wherein the primary to secondary leakage through the steam generators is determined via a radiochemistry sampling process, 3) Steam generator tube inspection activities wherein 100 percent of the repairs performed with the new process will be inspected during subsequent inservice inspections, 4) Inservice inspection reporting requirements wherein steam generator tubes, repaired with the new process, will be reported to the NRC in the annual report.

FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS, AND ACTIVITIES:

The Reactor Coolant System (RCS) is discussed in the DBNPS USAR Section 5.0, "Reactor Coolant System," and USAR Section 6.3, "Emergency Core Cooling System."

The RCS, in general, consists of the reactor vessel, two vertical once- through steam generators, four shaft-sealed reactor coolant pumps, an electrically heated pressurizer, and interconnecting piping. The system, located entirely within the Containment Vessel, is arranged in two heat transport loops, each with two reactor coolant pumps and one steam generator. Reactor coolant is transported through piping connecting the reactor vessel to the steam generators and flows downward through the steam generator tubes, transferring heat to the steam and water on the shell side of the steam generator. In each loop, the reactor coolant is returned to the reactor through two lines, each containing a reactor coolant pump.

The RCS performs the following functions which are important to safe plant operation:

- The RCS transfers heat from the core to the Steam Generators during steady state operation and for any design transient without exceeding core thermal limits.

- The RCS removes decay heat from the core via redundant components and features. The RCS is designed to be capable of natural circulation cooldown from normal operating temperature and pressure to conditions that permit operation of the Decay Heat Removal System.

- The RCS forms a barrier against the release of reactor coolant and radioactive material to the environs.

- The RCS transfers heat from the reactor core to containment during a loss of Steam Generator cooling with high RCS pressure, utilizing Make-up/High Pressure Injection (MU/HPI) Core Cooling.

The functions of the steam generators are to:

- Provide a pressure boundary between the reactor coolant and the secondary side fluid and to confine fission products and activation products within the reactor coolant system.

- Provide heat transfer capability and a heat sink to remove the reactor coolant heat produced during normal power operations.

- Provide normal and auxiliary feedwater flow paths and heat transfer capability for both normal and emergency cooldown.

- Supply steam for the auxiliary feed pumps' turbines for emergency cooling.

The repair roll method proposed in this LAR provides an additional method of repair for steam generator tubes that have sustained degradation in the tube within the steam generator upper tubesheet. The repair roll method will ensure that the area of degradation does not serve as a pressure boundary, thus permitting the tube to remain in service. Primary to secondary leakage surveillance activities provide assurance of steam generator tube integrity and that tubes have an adequate margin of safety to withstand the loads imposed during normal operation and postulated accidents. Steam generator tube inspection activities assure that the steam generator tubes are in satisfactory condition and that the progress of tube degradation is sufficiently slow to assure that the tubes will maintain their integrity during the operating period between inspections. The function of the reporting requirements for steam generator inservice inspections is to provide details of the steam generator tube degradation to the NRC.

EFFECTS ON SAFETY:

This LAR requests the addition of the repair process for the steam generator tubes called "Repair Roll" and a reduced limit of 150 GPD primary to secondary leakage through the tubes of any one steam generator.

The repair roll is a process whereby a new primary to secondary pressure boundary joint is established by hard rerolling the tube closer to the secondary face of the tubesheet. The new pressure boundary joint is established to remove the area of degradation from pressure boundary service (see the attached figure). The repair roll process was qualified for Inconel Alloy 600 material tubes in the upper tubesheets by Framatome Technologies Incorporated (FTI) for Babcock and Wilcox OTSGs. The DBNPS OTSG tubes are fabricated of Inconel Alloy 600 material. Framatome Technologies Incorporated Topical Report BAW-2303P, Revision 3, "OTSG Repair Roll Qualification Report," provides the basis for the acceptability of the repair roll process resulting in a tube-to-tubesheet joint equivalent to the original construction. BAW-2303P is provided as an attachment to this LAR. The report is proprietary to FTI and is so marked. Responses to previous NRC

questions regarding the repair roll process have been addressed by the Duke Power Company's letter to the NRC dated November 3, 1997, which resulted in Revision 3 to the Topical Report.

The repair roll qualification is based on establishing a mechanical joint, with leakage integrity, that is capable of carrying all normal operating and accident condition structural loads imposed on the steam generator tube without any support from the original tube expansion and tube seal weld. On the basis of a qualification program, a 1 inch roll length for the new joints was established to carry all structural loads and minimize potential leakage. The qualification program consisted of 1) preparing the mockup, 2) establishing tube loads for the qualification tests, and 3) performing verification tests and analyses.

The mockup consisted of a perforated metal block with eight inserted steam generator tubes that simulated the tube-to-tubesheet configuration in the field. The tubes were expanded into the mockup tubesheet using an expanding tool that had the same critical dimensions as the tool used in the field.

A Main Steam Line Break Accident was identified as the limiting design basis accident. The strength of the repair roll was demonstrated in accordance with the safety factor guidance of draft NRC Regulatory Guide (RG) 1.121, "Bases for plugging Degraded PWR Steam Generator Tubes" dated August 1976, by applying test pressure to the sample tubes in excess of 3 times normal operating differential pressure and 1.43 times Main Steam Line Break differential pressure. Regulatory Guide 1.121 recommends that the margin of safety against tube rupture under normal operating conditions should be equal to or greater than 3 at any tube location where defects have been detected. For postulated accidents, RG 1.121 recommends that the margin of safety against tube rupture be consistent with the margin of safety determined by the stress limits specified in NB-3225 of Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers. Normal structural loads imposed on the tube-to-tubesheet roll primarily are derived from the differential pressure between the primary and secondary sides of the tubes. Loadings from a postulated main steam line break accident were evaluated. In addition, cyclic loading from transients (e.g., startup/shutdown) was also considered in the qualification of the roll joints.

A series of tests and analyses was performed to establish the mechanical joint roll length and to verify the load carrying capabilities and leakage integrity of the repair roll joint (the repair roll also provides a mechanical seal for limiting leakage). The process verification testing included repair roll joint tensile strength and leakage testing. Thermal and fatigue cycling of the repair roll joint was followed by final tensile (ultimate) strength testing. Tests were performed to determine tube elongation rate and the effect of the repair roll process on tube axial loads.

Additionally, testing to determine tube springback rate from the repair roll process was performed to support analysis of tube radial contact stresses. The analyses evaluated plant operating and accident loads in addition to tubesheet bow effects.

Tubesheet bowing will cause the tubesheet bore diameter to increase or decrease during certain operating conditions due to the combined effects of primary to secondary pressure differential and thermal loads. An increase in diameter will decrease the contact stress between the repair roll joint and the tubesheet, which reduces the pullout strength. This effect on the strength of the repair roll joint was evaluated by FTI analytically and an exclusion zone was defined in BAW-2303P to ensure that the repair roll joint is installed only in locations where the effects of tubesheet bow do not reduce the joint strength below what is required to sustain all required loads. The addition of SR 4.4.5.4.a.7, referencing that the repair roll process of BAW-2303P will be used, will control the application of the exclusion zone.

The testing and analyses performed demonstrated that the repair roll process establishes a tube-to-tubesheet joint with leakage integrity capable of carrying all mechanical and thermal tube loads during normal operations and all postulated accidents. Based on the testing and analyses, and the fact that the roll is equivalent to original construction, there are no new safety issues associated with the repair roll process. Accordingly, a steam generator tube rupture is not more likely to occur in tubes repaired by the repair roll method. Additional details about the testing and analyses are contained in the attached proprietary FTI report.

The repair roll is typically installed using a manipulator and a tool head, monitored by a control system that tracks the position and monitors the torque of the roll expander. The roll expander is 1 1/2" long which provides a 1" long roll and a 1/4" long tapered transition on each end. The installation of a repair roll is a torque controlled process. The calibration of the torque controller and periodic inspection of the roller occurs during the installation of repair roll joints to ensure the installed repair rolls are equivalent to the samples tested during process qualification.

During the Tenth Refueling Outage in 1996, a portion of a tube was pulled from the upper tubesheet of one steam generator at the DBNPS and examined. No significant tubesheet crevice deposits were identified. This is consistent with the assumptions used in the FTI qualification process regarding crevice deposits affecting the leakage integrity of rerolled tubes.

As discussed in BAW-2303P, the repair process has no adverse effects on the function of the RCS or the steam generators. Steam generator tubes repaired with the repair roll process can remain in service as the repair roll joint is capable of carrying all structural loads imposed on the steam generator tube. Reactor coolant flow through the repaired tube is unaffected by the repair, and worst-case leakage through the repair joint is very low as reported in the FTI report.

After the steam generator tube repair rolls are completed, eddy current testing (ECT) is performed. This ECT examination confirms the proper tubes were expanded, verifies diametral expansion, and verifies the new roll expansion is free of degradation.

Alloy 600 tubes that have been cold worked by a process similar to the repair roll have shown themselves to be susceptible to stress corrosion cracking. Therefore, the new roll transitions may eventually exhibit defects. However, this type of defect is axial in nature, and readily detected by eddy current inspection during normal refueling outage inspections. The repair roll process specifies an eddy current examination to verify that the new roll joint is free of defects. This process, in conjunction with the requirement to inspect 100% of repair roll joints during future inservice inspections, will ensure that any future degradation will be detected prior to tube rupture or excessive leakage in this tube area.

The tubesheet material is expected to be unaffected by corrosion after installing a repair roll, even if defects currently exist. This is because tubesheet corrosion is based on the restricted flow area for primary water to interact with the tubesheet and the lack of oxygen in the primary system during normal operating conditions.

Based on the FTI qualification performed, there are no new safety issues associated with this repair roll. The proposed LCO 3.4.6.2.c reduction in the maximum allowable primary to secondary leakage through steam generators has no adverse effects on the function of the RCS or the steam generators. The reduction in leakage from 1 GPM (1440 GPD) total primary to secondary leakage to 150 GPD primary to secondary leakage through the tubes of any one steam generator provides defense-in-depth assurance of steam generator tube integrity and that tubes have an adequate margin of safety to withstand the loads imposed during normal operation and postulated accidents.

The addition of a SR for evaluating secondary water radiochemistry for determination of primary to secondary leakage through the steam generators is associated with the more restrictive LCO limit. The 72-hour determination frequency is similar to the 72 hour time requirement for performing a Reactor Coolant System water inventory balance. This SR provides for early detection of primary to secondary leakage through the steam generators and does not have an adverse effect on nuclear safety.

SIGNIFICANT HAZARDS CONSIDERATION:

The Nuclear Regulatory Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazard exists due to a proposed amendment to an Operating License for a facility. A proposed amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed changes would: (1) Not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) Not create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Not involve a significant reduction in a margin of safety. The Davis-Besse Nuclear Power Station has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, (DBNPS) Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because the proposed changes described for Surveillance Requirements (SR) 4.4.5.2.a.1, SR 4.4.5.4.a.4, SR 4.4.5.4.a.6, SR 4.4.5.4.a.7, SR 4.4.5.4.b, SR 4.4.5.4.a.9, SR 4.4.5.5.b.3, and Table 4.4-2 add a repair process defined as "repair roll" and redefine the pressure boundary joint for a tube repaired by the repair roll process. The application of the repair roll process is limited to repairs in the upper tube sheet. The new pressure boundary joint created by the repair roll process has been shown by testing and analysis to provide structural and leakage integrity equivalent to the original design and construction for all normal operating and accident conditions. Furthermore, the testing and analysis demonstrate the repair roll process creates no new adverse effects for the repaired tube and does not change the design or operating characteristics of the steam generators. Similarly, the design and operating characteristics of the systems interfacing with the steam generators are preserved by the repair roll process. Accordingly, tubes repaired by the repair roll process will not increase the probability of the tube rupture accident previously analyzed.

The proposed change to SR 4.4.5.3.c.1 and the proposed addition of SR 4.4.5.9 define additional required inspections for the primary system to secondary system joints created by the repair roll process. The addition of this inspection does not change any accident initiators and, therefore, does not increase the probability of an accident previously evaluated.

The proposed change to Limiting Condition for Operation (LCO) 3.4.6.2.c reduces the maximum allowed primary-to-secondary leakage through the steam generators from 1 gallon per minute (1440 GPD) to 150 GPD through any one steam generator. The reduction in allowed primary-to-secondary leakage does not change any accident initiators and, therefore, does not increase the probability of an accident previously evaluated.

The proposed additional requirements of SR 4.4.6.2.1.e describe the method and frequency that will be used for monitoring the reduced leakage limit. This additional monitoring of primary to secondary leakage through the steam generators does not change any accident initiators and, therefore, does not increase the probability of an accident previously evaluated.

The proposed changes to Bases B 3/4.4.5 add reference to the repair roll method and change the description of the allowed primary to secondary leakage through the steam generators to the reduced limit of 150 GPD through any one steam generator. It is noted that in Bases 3/4.4.5 the leakage limit established is defined as an inservice indicator of the structural integrity of the tubes. The reduction in the allowed primary to secondary leakage continues to provide inservice indication of tube structural integrity such that adequate margins of safety exist to withstand the loads imposed by normal operations and postulated accidents. Each of these changes to the

Bases does not change any accident initiators and, therefore, does not increase the probability of an accident previously evaluated.

The proposed changes to Bases 3/4.4.6.2 also change the description of the maximum allowed primary-to-secondary leakage to the lowered limit of 150 GPD through any one steam generator. The reduction of allowed primary-to-secondary leakage does not increase the probability of an accident previously evaluated.

The proposed changes to SR 4.4.5.2.a and SR 4.4.5.3.a are administrative changes and do not affect the probability of accidents previously evaluated.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes described for SR 4.4.5.2.a.1, SR 4.4.5.4.a.4, SR 4.4.5.4.a.6, SR 4.4.5.4.a.7, SR 4.4.5.4.b, SR 4.4.5.4.a.9, SR 4.4.5.5.b.3, and Table 4.4-2 add a repair process defined as "repair roll" and redefine the pressure boundary joint for a tube repaired by the repair roll process. The application of the repair roll process is limited to repairs in the upper tube sheet. The new pressure boundary joint created by the repair roll process has been shown by testing and analysis to provide structural and leakage integrity equivalent to the original design and construction for all normal operating and accident conditions. Furthermore, the testing and analysis demonstrate the repair roll process creates no new adverse effects for the repaired tube and does not change the design or operating characteristics of the steam generators. Similarly, the design and operating characteristics of the systems interfacing with the steam generators are preserved by the repair roll process. Accordingly, tubes repaired by the repair roll process will not increase the consequences of an accident previously analyzed. At worst, tubes repaired by the repair roll process will result in primary-to-secondary leakage. Should a tube leak occur, it would be bounded by the steam generator tube rupture accident consequences, which have been analyzed previously.

The proposed change to SR 4.4.5.3.c.1 and the proposed addition of SR 4.4.5.9 define additional required inspections for the primary system to secondary system joints created by the repair roll process. The addition of this inspection requirement does not increase the consequences of an accident previously evaluated.

The proposed change to LCO 3.4.6.2.c reduces the maximum allowed primary-to-secondary leakage through the steam generators from 1440 GPD to 150 GPD through any one steam generator. This change provides additional conservatism in the operation of the DBNPS and does not increase the consequences of an accident previously evaluated.

The proposed additional requirements of SR 4.4.6.2.1.e describe the method that will be used for monitoring the reduced leakage limit. This additional method of monitoring primary to secondary leakage through the steam generators does not change any accident and, therefore, does not increase the consequences of any accident previously evaluated.

The proposed changes to Bases B 3/4.4.5 add reference to the repair roll method and change the description of the allowed primary to secondary leakage through the steam generators to the reduced limit of 150 GPD through any one steam generator. It is noted that in Bases 3/4.4.5 the leakage limit established is defined as an inservice indicator of the structural integrity of the tubes. The reduction in the allowed primary to secondary leakage continues to provide inservice indication of tube structural integrity such that adequate margins of safety exist to withstand the loads imposed by normal operations and postulated accidents. These changes to the Bases do not change any accident and, therefore, will not increase the consequences of any accident previously evaluated.

The proposed changes to Bases 3/4.4.6.2 also change the description of the maximum allowed primary-to-secondary leakage to the lowered limit of 150 GPD through any one steam generator. The reduction of allowed primary-to-secondary leakage does not increase the consequences of any accident previously evaluated.

The changes to SR 4.4.5.2.a and SR 4.4.5.3.a are administrative changes and do not affect the consequences of accidents previously evaluated.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because there is no change in the operation of the steam generators or connecting systems with the repair roll process added by the proposed changes in SR 4.4.5.2.a.1, SR 4.4.5.4.a.4, SR 4.4.5.4.a.6, SR 4.4.5.4.a.7, SR 4.4.5.4.a.9, SR 4.4.5.4.b, SR 4.4.5.5.b.3 and Table 4.4-2. The physical changes in the steam generators associated with the repair roll process have been evaluated and do not create the possibility for a new or different kind of accident from any accident previously evaluated, i.e., the physical change in the steam generators is limited to the location of the primary to secondary boundary within the tubesheet and does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The reduction in maximum allowed primary-to-secondary leakage defined by the proposed change to LCO 3.4.6.2.c does not create the possibility of a new or different kind of accident from any previously evaluated accident. The additional testing of tubes repaired by the repair roll process as required by the proposed change to SR 4.4.5.3.c.1 and the addition of SR 4.4.5.9 does not create the possibility of a new or different kind of accident from any previously evaluated accident. Similarly, the monitoring of primary to secondary leakage

as specified in the proposed SR 4.4.6.2.1.e does not create the possibility of a new or different kind of accident from any previously evaluated accident.

The proposed changes to Bases 3/4.4.5 and 3/4.4.6.2 reflect the changes proposed to their associated LCOs and SRs, and are not involved with any accident. The changes made to SR 4.4.5.2.a and SR 4.4.5.3 a are administrative changes and do not create the possibility of new or different kinds of accidents from any accident previously evaluated.

3. Not involve a significant reduction in a margin of safety because all of the protective boundaries of the steam generator are maintained equivalent to the original design and construction with tubes repaired by the repair roll process. Furthermore, tubes with primary system to secondary system boundary joints created by the repair roll have been shown by testing and analysis to satisfy all structural, leakage, and heat transfer requirements.

   The additional testing of tubes repaired by the repair roll process provides continuing inservice monitoring of these tubes such that inservice degradation of tubes repaired by the repair roll process will be detected. Therefore, the changes to SR 4.4.5.2.a.1, SR 4.4.5.4.a.4, SR 4.4.5.4.a.6, SR 4.4.5.4.a.7, SR 4.4.5.4.b, SR 4.4.5.5.b.3 and Table 4.4-2 to add repair roll as a repair process do not reduce a margin of safety. Similarly, the proposed change to SR 4.4.5.4.a.9 to redefine the pressure boundary for a tube with a repair roll is based upon eddy current testing demonstrating the adequacy of the repair roll to provide this pressure boundary and maintain the present margin of safety.

   The proposed reduction of allowed primary to secondary leakage, as defined in the changes to LCO 3.4.6.2.c, constitutes additional conservatism in the operation of the DBNPS and does not reduce a margin of safety. Similarly, the additional testing and monitoring defined in the changed SR 4.4.5.3.c.1 and the proposed SR 4.4.5.9 and SR 4.4.6.2.1.e constitute additional conservatism in the operation of the DBNPS and do not reduce a margin of safety.

   The proposed changes to Bases 3/4.4.5 and 3/4.4.6.2 reflect the changes proposed to their associated LCOs and SRs, and do not reduce a margin of safety.

   The changes to SR 4.4.5.2.a and SR 4.4.5.3.a are administrative changes and do not reduce the margin of safety.

CONCLUSION:

On the basis of the above, the Davis-Besse Nuclear Power Station has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory

Commission, this License Amendment Request does not constitute an unreviewed safety question.

ATTACHMENTS:

Attachment 1 is the proposed marked-up changes for the Operating License.

Attachment 2 is the proprietary Framatome Technologies Topical Report BAW-2303P, Revision 3, "OTSG Repair Roll Qualification Report."

REFERENCES:

1. USAR Section 5.0, "Reactor Coolant System," through Revision 20.

2. USAR Section 6.3, "Emergency Core Cooling System," through Revision 20.

3. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 215.

4. DBNPS System Description, SD-041 R1, "Steam Generator."

5. DBNPS System Description, SD-39A, "Reactor Coolant System."

6. Framatome Technologies Topical Report, BAW-2303P Revision 3, "OTSG Repair Roll Qualification Report," dated October 1997.

7. Draft NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," dated August 1976.

8. Duke Energy Oconee Nuclear Station Units 1, 2 and 3 Operating Licenses Amendments Number 227, 227, and 224, dated November 21, 1997.

9. Duke Power Company letter to the NRC, "Oconee Nuclear Station Docket Nos. 50-269, -270, -287. Request for Technical Specification Amendment Steam Generator Tubing Surveillance," dated November 3, 1997.

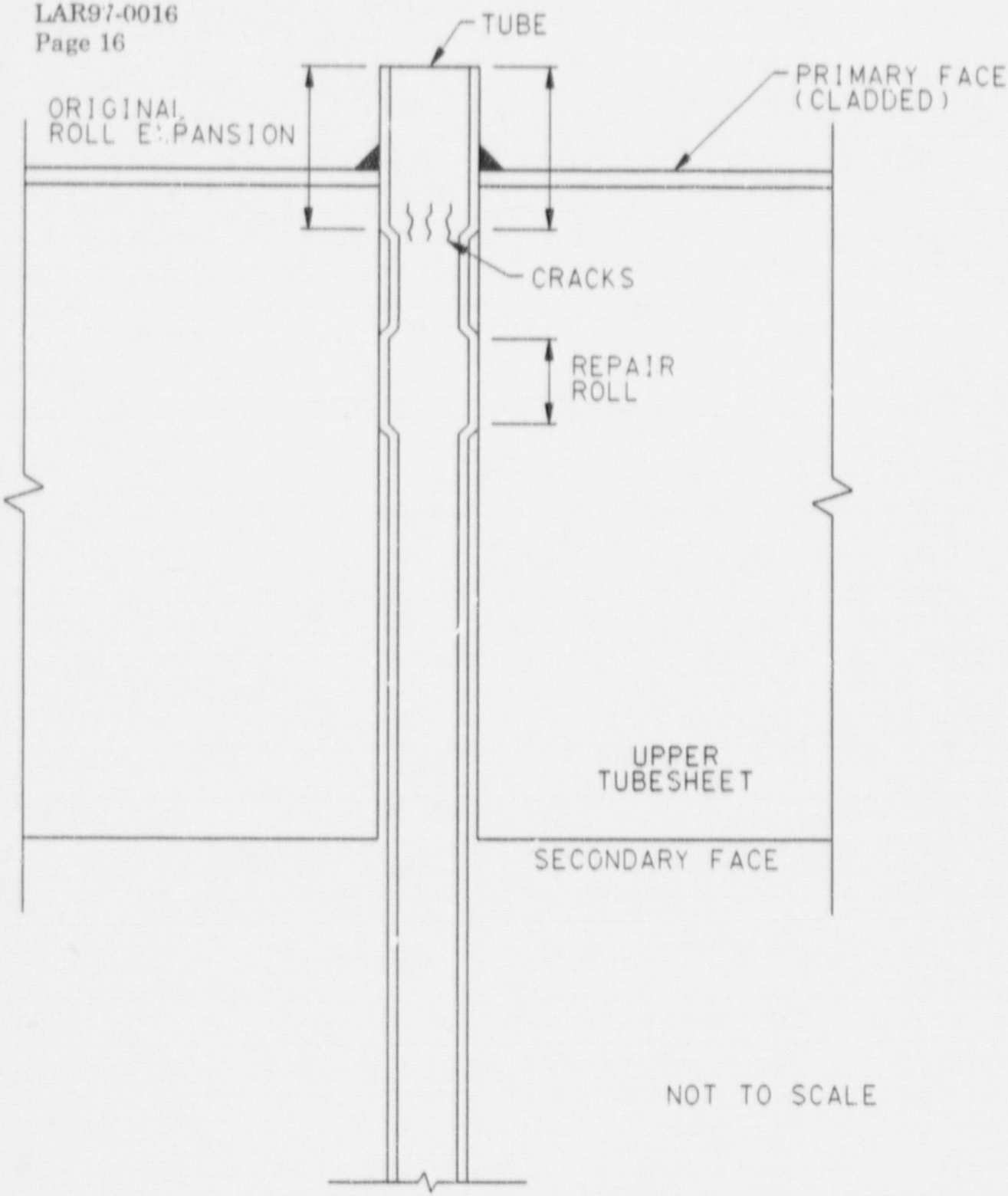

FIGURE
TUBE REPAIR ROLL