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Note: The enclosure to this letter contains proprietary information

Docket Number 50-346

License Number NPF-3

Serial Number 2634

January 25, 2000

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555-0001

Subject: Proposed Modification to the Davis-Besse Nuclear Power Station Operating License NPF-3, Appendix A Technical Specification (TS) 3/4.4.5, Steam Generators, to Modify Single Repair Roll Requirements and Allow the Use of the Double Repair Roll Process for Steam Generator Tubes Within the Upper Tubesheet (LAR No. 99-0001; TAC No. MA7829)

Ladies and Gentlemen:

Enclosed is an application for an amendment to the Davis-Besse Nuclear Power Station (DBNPS) Unit Number 1, Facility Operating License NPF-3, Appendix A, Technical Specifications. The proposed change involves the Once-Through Steam Generator (OTSG) tube repair roll process. The repair roll process is used for the repair of OTSG tubes with defects within the upper tubesheet. The proposed change involves Technical Specification (TS) 3/4.4.5, Reactor Coolant System - Steam Generators, and its associated Bases.

The proposed revision would establish updated regions in the OTSG upper tubesheet where repair roll installation is prohibited by adopting the requirements contained in the enclosed proprietary Framatome Technologies Incorporated Topical Report BAW-10236P, Revision 0, "Addendum for Davis-Besse Repair Roll UTS Exclusion Zones." A non-proprietary version of the topical report is also enclosed. The proposed revision would also allow the use of two overlapping repair rolls to produce a double repair roll. Similar changes to Technical Specification requirements have been reviewed by the NRC in approving License Amendment No. 180 to Facility Operating License No. DPR-72 for Crystal River Unit 3, dated June 28, 1999 (TAC No. MA3592).

T007

The Twelfth Refueling Outage (12RFO) for the DBNPS is presently scheduled to commence on April 1, 2000, which will include inservice inspection of the OTSGs. The FirstEnergy Nuclear Operating Company (FENOC) requests NRC approval of this OTSG repair method so that it may be available for use, should it be needed. Accordingly, the FENOC requests that the NRC approve and issue these TS changes by March 31, 2000.

Should you have any questions or require additional information, please contact Mr. James L. Freels, Manager - Regulatory Affairs, at (419) 321-8466.

Very truly yours,

MAR/laj

Enclosures

cc: J. E. Dyer, Regional Administrator, NRC Region III
D. V. Pickett, DB-1 NRC/NRR Senior Project Manager
J. R. Williams, Chief of Staff, Ohio Emergency Management Agency, State of Ohio (NRC Liaison)
K. S. Zellers, DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board

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 2634) concern:

> Appendix A, Technical Specification Sections (TS) 3/4.4.5, Reactor Coolant System - Steam Generators, and its associated Bases.

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information, and belief.

By: Guy G. Campbell

Guy G. Campbell, Vice President - Nuclear

Sworn to and subscribed before me this 25th day of January, 2000

Laura A. Jennison

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, and its associated Bases.

A. Time Required to Implement: This change is to be implemented within 120 days after the NRC issuance of the License Amendment.

B. Reason for Change (License Amendment Request Number 99-0001):

   This application proposes to revise the TS Surveillance Requirement (SR) 4.4.5.4.a.7 and Bases 3/4.4.5 for steam generator tube repair. The proposed changes would modify the repair roll process to update exclusion zones and allow the use of the double repair roll for the repair of OTSG tubes with defects within the upper tubesheet. The "repair roll" discussed in existing SRs 4.4.5.2.a.1, 4.4.5.3.c.1, 4.4.5.4.a.4, 4.4.5.4.a.6, 4.4.5.4.a.7, 4.4.5.4.a.9, 4.4.5.4.b, 4.4.5.5.b.3, and 4.4.5.9 and TS Table 4.4-2 would include the double repair roll in addition to the single repair roll.

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

D. Proprietary Framatome Technologies Incorporated Topical Report BAW-10236P, Revision 0, "Addendum for Davis-Besse Repair Roll UTS Exclusion Zones," dated December 1999: See Attachment 2.

Docket Number 50-346
License Number NPF-3
Serial Number 2634
Attachment 1

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR LICENSE AMENDMENT REQUEST NO. 99-0001

(24 Pages Follow)

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR LICENSE AMENDMENT REQUEST NUMBER 99-0001

TITLE:

Proposed Modification to the Davis-Besse Nuclear Power Station (DBNPS) Operating License NPF-3, Appendix A Technical Specification (TS) 3/4.4.5, Steam Generators, to Modify Single Repair Roll Requirements and Allow the Use of the Double 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 during the OTSG fabrication and are about 1 to 2 inches in axial length, expanded into the tubesheet. These tube-to-tubesheet rolls hold the tubes axially in 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.

It is desirable to repair degraded OTSG tubes, if possible, and retain the tubes in service for providing heat transfer, rather than plugging the tubes and removing them from service. The current DBNPS Technical Specification 3/4.4.5, Steam Generators, Surveillance Requirement (SR) 4.4.5.4.a.7 allows "rolling" as a repair method for OTSG tubes. Tube repair rolling is a repair process for OTSG degraded tubes only when 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). Under the current DBNPS TS requirements, the OTSG tubes with defects in the upper tubesheet area may be repaired by performing a single one inch long tube roll expansion (single roll) in accordance with Framatome Technologies Incorporated (FTI) Topical Report BAW-2303P, Revision 3, "OTSG Repair Roll Qualification Report," dated October 1997, (previously submitted to the NRC by DBNPS Letter Serial No. 2512, dated February 26, 1998).

This License Amendment Request (LAR) proposes the revision of the DBNPS Technical Specifications to modify the steam generator tube repair roll requirements. The proposed revision would modify SR 4.4.5.4.a.7 to require that repair rolls be performed in accordance with attached FTI Topical Report BAW-10236P, Revision 0, "Addendum for Davis-Besse Repair Roll UTS Exclusion Zones," which supplements the requirements of Topical Report BAW-2303P, Revision 3. Topical Report BAW-10236P contains updated limiting tensile tube loads and defines new exclusion zones within the steam generator in which the application of the repair roll is prohibited.

The proposed revision would also eliminate the restriction on only using a single roll and permit the use of two overlapping 1-inch tube roll expansions to produce a 1-5/8 inches long repair roll ("double roll"). The double repair roll provides a tube-to-tubesheet joint that is stronger than the single repair roll. The proposed revision would require the double repair roll process be performed in accordance with Topical Reports BAW-2303P, Revision 3, as supplemented by Topical Report BAW-10236P, Revision 0. Topical Report BAW-10236P also defines steam generator tube exclusion zones where application of the single or double repair rolls is prohibited, based upon the tensile load on the tubes. A location is excluded if the repair roll load carrying capability is predicted to be less than 110% of the limiting transient tube load. Technical Specification Bases 3/4.4.5 is also proposed for revision to make it consistent with the proposed revised SR 4.4.5.4.a.7.

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

SR 4.4.5.4.a.7 currently states, in part:

> 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.

The proposed revised SR 4.4.5.4.a.7 portion would read as follows:

> The repair roll process will only be used to repair tubes with degradation or defects in the upper tubesheet area. The repair roll process (either single roll or double roll) may be performed only once per steam generator tube. The new roll area must be free of imperfections and 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, as supplemented by Topical Report BAW-10236P, Revision 0.

These proposed revisions would permit the use of a double roll as a tube repair option and require the use of the updated postulated tube loads contained in FTI Topical Report BAW-10236P when determining repair requirements.

Bases Section 3/4.4.5, "Steam Generators," currently contains the following paragraph:

> 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.

The proposed revised paragraph would read as follows:

> The repair roll process (either single or double roll) may be performed only once per steam generator tube using the methodology described in the Topical Reports BAW-2303P, Revision 3, and BAW-10236P, Revision 0. Topical Report BAW-10236P, Revision 0, replaced portions of Topical Report BAW-2303P, Revision 3, regarding the repair roll process. The new roll area must be free of imperfections and 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.

This proposed revision is required to maintain Bases Section 3/4.4.5 consistent with the revised SR 4.4.5.4.a.7.

The use of the single and double repair roll processes for a Babcock and Wilcox OTSG have been previously reviewed by the NRC in approving License Amendment No. 180 to Facility Operating License No. DPR-72 for Crystal River Unit 3, dated June 28, 1999 (TAC No. MA3592).

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED:

The following systems and components are affected by the proposed changes to TS SR 4.4.5.4.a.7 and Bases 3/4.4.5: the Reactor Coolant System, Steam Generators, and the Steam Generator tubes inside of the tubesheet.

The following activity is affected: 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.

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 pump turbines for emergency cooling.

The repair roll method being modified in this LAR provides a method of repair for steam generator tubes that have sustained degradation of the tube within the steam generator upper tubesheet. 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).

EFFECTS ON SAFETY:

General Design Criteria 14 of 10 CFR 50, Appendix A, requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of a gross rupture. A portion of the RCPB is maintained by the OTSG tubes which have experienced various levels of degradation.

This LAR requests that the repair requirements of TS 4.4.5.4.a.7 incorporate the repair roll process of Framatome Technologies Incorporated (FTI) Topical Report BAW-10236P, Revision

0. In addition, this LAR requests the addition of the double repair roll as an acceptable method for OTSG tube repair within the tubesheet region.

The DBNPS Technical Specifications currently permit the use of a single one-inch long tube roll expansion as a repair technique for tubes with degraded roll joints in the upper tubesheet. FTI Topical Report BAW-2303P, Revision 3, "OTSG Repair Roll Qualification Report," provides the current basis for the use of the single repair roll process. FTI Topical Report BAW-2303P, Revision 3, defines regions of the upper tubesheet where installation of a single repair roll is not an acceptable repair process. These exclusion zones were established for regions where the tube load exceeded the repair roll strength. Recently FTI redefined the exclusion zones for the application of single repair rolls and also defined the exclusion zones for the application of the double repair roll. A location is excluded if the repair roll load carrying capability is predicted to be less than 110% of the limiting transient tube load.

FTI Topical Report BAW-10236P, Revision 0, was prepared to update the repair roll analysis for the DBNPS. Topical Report BAW-10236P is an addendum to Topical Report BAW-2303P that reevaluated the load carrying capabilities of repair rolls and established new exclusion areas for single repair roll installation. Topical Report BAW-10236P evaluates the Main Steam Line Break (MSLB) and a Small Break Loss of Coolant Accident (SBLOCA) conditions against the load carrying capability of a single repair roll. The exclusion zones for single repair roll installation established in Topical Report BAW-10236P are more limiting than those established in BAW-2303P and will be adopted.

These exclusion zone requirements imposed by BAW-10236P, Revision 0, supplement the requirements of BAW-2303P, Revision 3. FTI Topical Report BAW-10236P provides additional restrictions on the tube zones available for repair roll installation. Therefore, the adoption of the BAW-10236P, Revision 0, requirements has no adverse effect on nuclear safety.

The DBNPS Technical Specifications currently limit the repair roll process to a single one-inch long expansion. Currently, tubes found with defects in the upper tubesheet single repair roll exclusion zone have to be plugged or repaired by other NRC approved methods. This proposed revision would allow the use of two overlapping one-inch repair rolls to produce a 1-5/8 inch long double repair roll. The double repair roll process would use the same roll expander used in the single repair roll process. Double repair roll joints were qualified in FTI Topical Report BAW-2303P, Revision 3.

The repair roll methodology described in Topical Report BAW-2303P, Revision 3, was qualified in accordance with the guidance of draft NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," dated August 1976. A series of tests and analyses were performed to establish the capabilities and leakage integrity of the double repair roll joint. The process verification testing included tensile strength and leakage testing, and the determination of the effects of thermal and fatigue cycling of the repair roll joint. Tests were also performed to determine tube elongation rate and the effect of the double repair roll process on tube axial loads. As described in Topical Report BAW-10236P, under the combination of primary to secondary pressure differential and thermal loads, tubesheet bowing will cause the bore diameter around the tubes to change. An increase in diameter will decrease the contact stress between the repair roll joint and the tubesheet, which reduces the force by which the tube is held by the tubesheet. This effect on the double repair roll holding capacity was evaluated by FTI (utilizing axial tube loads associated with the most limiting OTSG tensile loads). Based on the effects of tubesheet bow that could reduce the joint strength below what is required to sustain all required loads, BAW-10236P, Revision 0, defines the OTSG tube zones where the double repair roll methodology may be implemented.

The testing and analyses performed demonstrated that the overlapping 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 except in the double repair roll OTSG tube exclusion zones. 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 double repair roll process. Accordingly, a steam generator tube rupture is not more likely to occur in tubes repaired by the double repair roll method. Additional details about the testing and analyses are contained in BAW-2303P, Revision 3, and BAW-10236P.

Installation of the repair rolls will be in accordance with Topical Report BAW-2303P, Revision 3, as supplemented by BAW-10236P, Revision 0. Topical Report BAW-2303P, Revision 3, Section 3.1, described the second roll as overlapping the original fabrication roll. However, in the case of the original fabricated roll being in a degraded state, it is necessary that the second roll be applied in a non-degraded area to best-assure that the completed double roll is free of any degradation. Accordingly, repair rolling over the original fabrication roll is now explicitly disallowed by Topical Report BAW-10236P. Inspection of the repair roll area prior to installation ensures the new roll area is free of degradation and imperfections. Inspection of the repair roll following installation verifies the repair roll length and ensures the new tube-to-tubesheet joint is free of anomalies.

The tubes which require the double repair roll are known to be susceptible to stress corrosion cracking. The repair roll may eventually exhibit defects as some of the original roll transitions have. For this reason, the repair roll joint and transitions for all tubes that have been repaired by the repair roll process will be inspected during each steam generator inspection in accordance with existing SR 4.4.5.9.

As discussed in Topical Report BAW-2303P, Revision 3, the repair process will have no adverse effect on the function of the RCS or the steam generators. Steam generator tubes repaired with the repair roll process can remain in service because 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 projected to be very low as discussed in the Topical Report BAW-2303P, Revision 3.

Based on the analyses performed and discussed above, there are no safety issues associated with this repair roll. Therefore, the proposed Technical Specification revisions will have no adverse effects on the nuclear safety of the RCS or the steam generators.

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 testing and analysis have shown the proposed repair roll process to be added to Surveillance Requirement (SR) 4.4.5.4.a.7 ensures the new pressure boundary joint created by the repair roll process provides structural and leakage integrity equivalent to the original design and construction for all normal operating and accident conditions. The proposed repair roll process does not alter the design or operating characteristics of the steam generators or systems interfacing with the steam generators. Therefore, the proposed changes to SR 4.4.5.4.a.7 will not increase the probability of a previously evaluated accident.

The proposed change to Bases 3/4.4.5 reflects the changes proposed to its associated SR, and does not involve an increase in the probability of an accident previously evaluated.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed repair roll process to be added to Surveillance Requirement (SR) 4.4.5.4.a.7 ensures the new pressure boundary joint created by the repair roll process provides structural and leakage integrity equivalent to the original design and construction for all accident conditions. Should a repaired tube fail, the radiological consequences would be bounded by the existing Steam Generator Tube Rupture analysis.

The proposed change to Bases 3/4.4.5 reflects the changes proposed to its associated SR, and does not involve an increase to the consequences of an accident previously evaluated.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because there will be no change in the operation of the steam generators or connecting systems as a result of the repair roll process added by the proposed changes to SR 4.4.5.4.a.7. 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. Furthermore, the repair roll process installs a pressure boundary joint equivalent to that of the original fabrication.

Accordingly, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to Bases 3/4.4.5 reflects the changes proposed to its associated SR, and does not create the possibility of any new or different kind of accident.

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 under existing SR 4.4.5.9 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 4.4.5.4.a.7 to modify the repair process do not reduce the margin of safety.

   The proposed change to Bases 3/4.4.5 reflects the changes proposed to its associated SR, and does 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 contains the proposed marked-up changes for the Operating License.

Attachment 2 is the proprietary Framatome Technologies Incorporated Topical Report BAW-10236P, Revision 0, "Addendum for Davis-Besse UTS Exclusion Zones."

REFERENCES:

1. USAR Section 5.0, "Reactor Coolant System," through Revision 21.

2. USAR Section 6.3, "Emergency Core Cooling System," through Revision 21.

3. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 234.

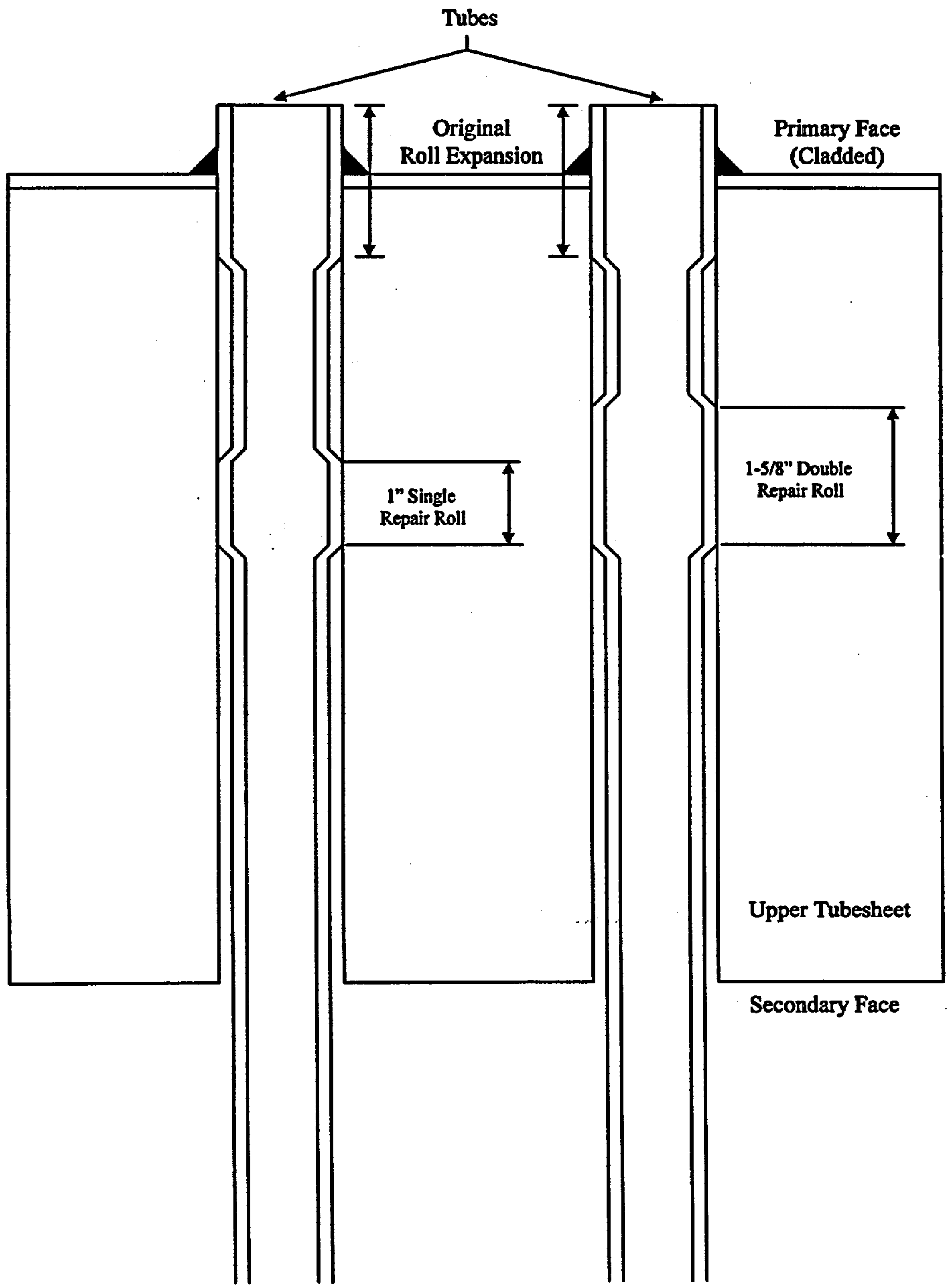
4. DBNPS System Description, SD-041 R1, "Steam Generator."

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

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

7. Framatome Technologies Incorporated Topical Report BAW-10236P, Revision 0, "Addendum for Davis-Besse UTS Exclusion Zones," dated December 1999.

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



Not to Scale

Figure
Tube Repair Rolls

INFORMATION ONLY

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each Steam Generator shall be OPERABLE with a minimum water level of 18 inches and the maximum specified below as applicable:

MODES 1 and 2:

a. The acceptable operating region of Figure 3.4-5.

MODE 3*:

b. 50 inches Startup Range with the SFRCS Low Pressure Trip bypassed and one or both Main Feedwater Pump(s) capable of supplying Feedwater to any Steam Generator.

c. 96 percent Operate Range with:

1. The SFRCS Low Pressure Trip active.

   Or

2. The SFRCS Low Pressure Trip bypassed and both Main Feedwater Pumps incapable of supplying Feedwater to the Steam Generators.

MODE 4:

d. 625 inches Full Range Level

APPLICABILITY: MODES 1, 2, 3, and 4, as above.

ACTION:

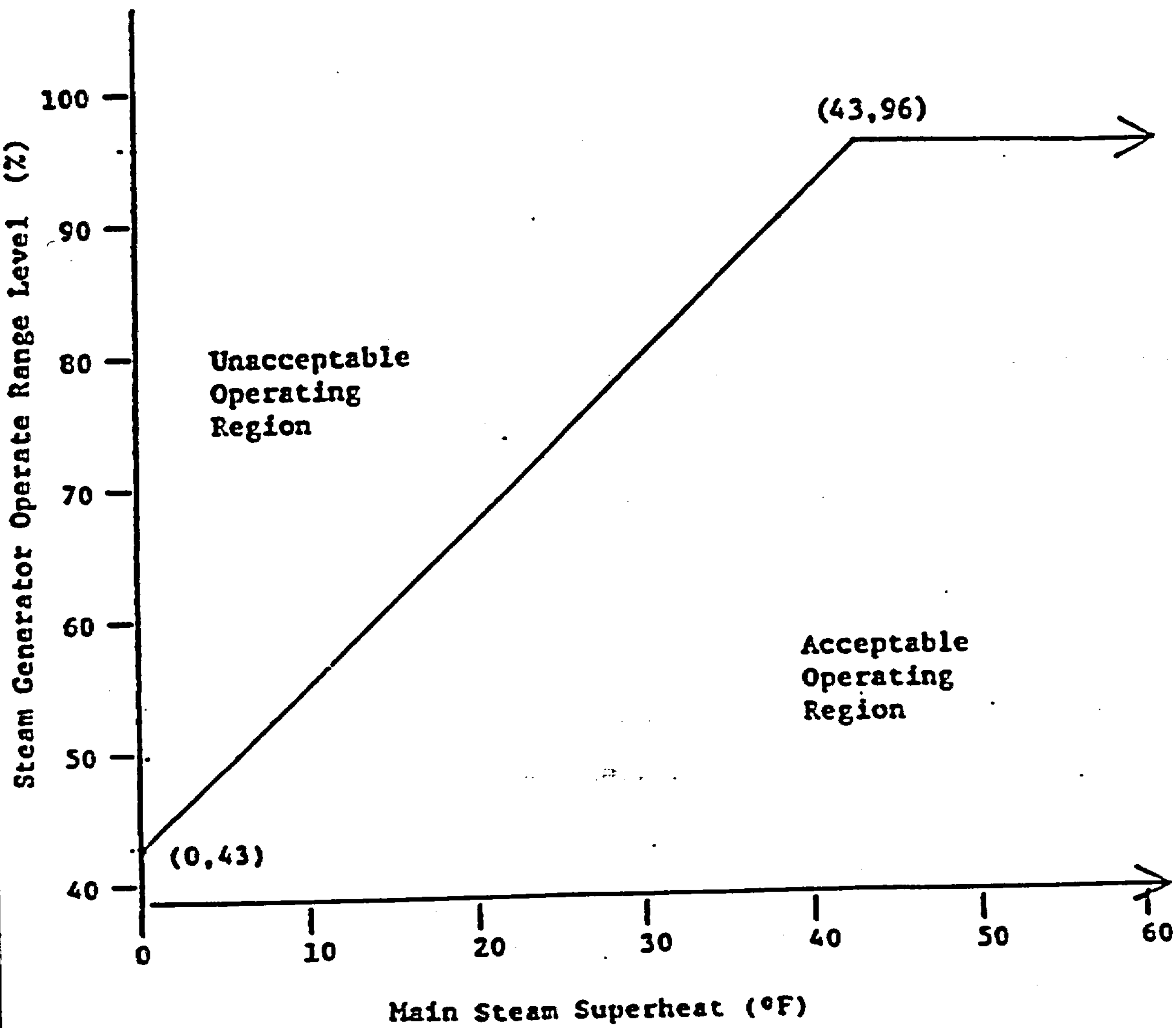
a. With one or more steam generators inoperable due to steam generator tube imperfections, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

b. With one or more steam generators inoperable due to the water level being outside the limits, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

*Establish adequate SHUTDOWN MARGIN to ensure the reactor will stay subcritical during a MODE 3 Main Steam Line Break.

INFORMATION ONLY

Figure 3.4-5
Maximum Allowable Steam Generator Level
in MODES 1 and 2



INFORMATION ONLY

REACTOR COOLANT SYSTEM

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4-4.1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. The first sample inspection during each inservice inspection of each steam generator shall include:

   1. All tubes or tube sleeves that previously had detectable wall penetrations (> 20%) that have not been plugged or repaired by repair roll or sleeving in the affected area. (Tubes repaired by sleeving or repair roll remain available for random selection).

   2. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.

INFORMATION ONLY

SURVEILLANCE REQUIREMENTS (Continued)

3. A tube inspection (pursuant to Specification 4.4.5.4.a.9) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

b. Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. No credit will be taken for these tubes in meeting minimum sample size requirements.

1. Group A-1: Tubes within one, two or three rows of the open inspection lane.

2. Group A-2: Tubes having a drilled opening in the 15th support plate.

3. Group A-3: Tubes included in the rectangle bounded by rows 62 and 90 and by tubes 58 and 76, excluding tubes included in Group A-1.*

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to less than a full tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

* Tubes in Group A-3 shall not be excluded after completion of the fifth refueling outage.

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Notes: (1) In all inspections, previously degraded tubes must exhibit significant (> 10%) further wall penetrations to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 4.4.5.2.b, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections for a given group* of tubes following service under all volatile treatment (AVT) conditions fall into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval for that group may be extended to a maximum of 40 months.

b. If the results of the inservice inspection of a steam generator performed in accordance with Table 4.4-2 at 40 month intervals for a given group* of tubes fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 20 calendar months after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 4.4.5.3a and the interval can be extended to 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to tube sheet welds) in excess of the limits of Specification 3.4.6.2.

   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 of the new roll areas in the unaffected steam generator.

*A group of tubes means:
(a) All tubes inspected pursuant to 4.4.5.2.b, or
(b) All tubes in a steam generator less those inspected pursuant to 4.4.5.2.b.

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2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

d. The provisions of Specification 4.0.2 are not applicable.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Tubing or Tube means that portion of the tube or tube sleeve which forms the primary system to secondary system boundary.

2. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

3. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

4. Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation that has not been repaired by repair roll or sleeving in the affected area.

5. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

6. Defect means an imperfection of such severity that it exceeds the repair limit. A defective tube is a tube containing a defect that has not been repaired by repair roll or sleeving in the affected area or a sleeved tube that has a defect in the sleeve.

7. Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by repair roll or sleeving in the affected area because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness. The Babcock and Wilcox process described in Topical Report BAW-2120P will be used for sleeving.

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(Continued) 7. The repair roll process will only be used to repair tubes with degradation or defects in the upper tubesheet area. The repair roll process (either single roll or double roll) may ~~will~~ be performed only once per steam generator tube ~~using a 1 inch reroll length~~. The new roll area must be free of imperfections and 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, as supplemented by Topical Report BAW-10236P, Revision 0.

8. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

9. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. The previously existing tube and tube roll, above the new roll area in the upper or lower tube sheet, can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

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10. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair by repair roll or sleeving in the affected areas all tubes exceeding the repair limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be submitted on an annual basis in a report for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged, sleeved or repair rolled.

c. Results of steam generator tube inspections which fall into Category C-3 and require notification of the Commission shall be reported prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

4.4.5.6 The steam generator shall be demonstrated OPERABLE by verifying steam generator level to be within limits at least once per 12 hours.

4.4.5.7 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 peripheral tubes in the vicinity of the secured internal auxiliary feedwater header. This testing shall only be required on the steam generator selected for inspection, and the test shall require inspection only between

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the upper tube sheet and the 15th tube support plate. The tubes selected for inspection shall represent the entire circumference of the steam generator and shall total at least 150 peripheral tubes.

4.4.5.8 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 steam generator through the auxiliary feedwater injection penetrations.

These inspections of the secured internal auxiliary feedwater header, header to shroud attachment welds, and the external header thermal sleeves shall be performed during the third period of each ten-year Inservice Inspection Interval (ISI).

4.4.5.9 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.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G. (1)	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair by repair rolling or sleeving defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or repair by repair rolling or sleeving defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair by repair rolling or sleeving defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug or repair by repair rolling or sleeving defective tubes and inspect 2S tubes in each other S.G. Report to the NRC prior to resumption of plant operation.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair by repair rolling or sleeving defective tubes. Report to the NRC prior to resumption of plant operation.	N/A	N/A

(1) $S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

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3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and pilot operated relief valve against water relief.

The low level limit is based on providing enough water volume to prevent a reactor coolant system low pressure condition that would actuate the Reactor Protection System or the Safety Feature Actuation System. The high level limit is based on providing enough steam volume to prevent a pressurizer high level as a result of any transient.

The pilot operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the pilot operated relief valve minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken. A process equivalent to the inspection method described in Topical Report BAW-2120P will be used for inservice inspection of steam generator tube sleeves. This inspection will provide ensurance of RCS integrity.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 150 GPD through any one steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal

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operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 150 GPD can be detected by monitoring the secondary coolant. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired by repair rolling or sleeving in the affected areas.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. As described in Topical Report BAW-212OP, degradation as small as 20% through wall can be detected in all areas of a tube sleeve except for the roll expanded areas and the sleeve end, where the limit of detectability is 40% through wall. Tubes with imperfections exceeding the repair limit of 40% of the nominal wall thickness will be plugged or repaired by repair rolling or sleeving the affected areas. Davis-Besse will evaluate, and as appropriate implement, better testing methods which are developed and validated for commercial use so as to enable detection of degradation as small as 20% through wall without exception. Until such time as 20% penetration can be detected in the roll expanded areas and the sleeve end, inspection results will be compared to those obtained during the baseline sleeved tube inspection.

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 (either single or double roll) may ~~will~~ be performed only once per steam generator tube using ~~a 1 inch reroll length as~~ the methodology described in the Topical Reports BAW-2303P, Revision 3, and BAW-10236P, Revision 0. Topical Report BAW-10236P, Revision 0, replaced portions of Topical Report BAW-2303P, Revision 3, regarding the repair roll process. ~~Thus, multiple applications of the rerolling process to any individual tube is not acceptable.~~ The new roll area must be free of imperfections and 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.~~

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Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results shall be reported to the Commission prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

The steam generator water level limits are consistent with the initial assumptions in the USAR. While in MODE 3, examples of Main Feedwater Pumps that are incapable of supplying feedwater to the Steam Generators are tripped pumps or a manual valve closed in the discharge flowpath. The reactivity requirements to ensure adequate SHUTDOWN MARGIN are provided in plant operating procedures.

Docket Number 50-346
License Number NPF-3
Serial Number 2634
Attachment 2

PROPRIETARY
FRAMATOME TECHNOLOGIES INCORPORATED
TOPICAL REPORT BAW-10236P, REVISION 0
"Addendum for Davis-Besse Repair Roll UTS Exclusion Zones "
DECEMBER 1999

AFFIDAVIT OF JOSEPH J. KELLY

A. My name is Joseph J. Kelly. I am Manager of B&W Owners Group Services for Framatome Technologies, Inc. (FTI), and as such, I am authorized to execute this Affidavit.

B. I am familiar with the criteria applied by FTI to determine whether certain information of FTI is proprietary and I am familiar with the procedures established within FTI to ensure the proper application of these criteria.

C. In determining whether an FTI document is to be classified as proprietary information, an initial determination is made by the Unit Manager, who is responsible for originating the document, as to whether it falls within the criteria set forth in Paragraph D hereof. If the information falls within any one of these criteria, it is classified as proprietary by the originating Unit Manager. This initial determination is reviewed by the cognizant Section Manager. If the document is designated as proprietary, it is reviewed again by me to assure that the regulatory requirements of 10 CFR Section 2.790 are met.

D. The following information is provided to demonstrate that the provisions of 10 CFR Section 2.790 of the Commission's regulations have been considered:

(i) The information has been held in confidence by FTI. Copies of the document are clearly identified as proprietary. In addition, whenever FTI transmits the information to a customer, customer's agent, potential customer or regulatory agency, the transmittal requests the recipient to hold the information as proprietary. Also, in order to strictly limit any potential or actual customer's use of proprietary information, the substance of the following provision is included in all agreements entered into by FTI, and an equivalent version of the proprietary provision is included in all of FTI's proposals:

"Any proprietary information concerning Company's or its Supplier's products or manufacturing processes which is so designated by Company or its Suppliers and disclosed to Purchaser incident to the performance of such contract shall remain the property of Company or its Suppliers and is disclosed in confidence, and Purchaser shall not publish or otherwise disclose it to others without the written approval of Company, and no rights, implied or otherwise, are granted to produce or have produced any products or to practice or cause to be practiced any manufacturing processes covered thereby.

Notwithstanding the above, Purchaser may provide the NRC or any other regulatory agency with any such proprietary information as the NRC or such other agency may require; provided, however, that Purchaser shall first give Company written notice of such proposed disclosure and Company shall have the right to amend such proprietary information so as to make it non-proprietary. In the event that Company cannot amend such proprietary information, Purchaser shall prior to disclosing such information, use its best efforts to obtain a commitment from NRC or such other agency to have such information withheld from public inspection.

Company shall be given the right to participate in pursuit of such confidential treatment."

(ii) The following criteria are customarily applied by FTI in a rational decision process to determine whether the information should be classified as proprietary. Information may be classified as proprietary if one or more of the following criteria are met:

a. Information reveals cost or price information, commercial strategies, production capabilities, or budget levels of FTI, its customers or suppliers.

b. The information reveals data or material concerning FTI research or development plans or programs of present or potential competitive advantage to FTI.

c. The use of the information by a competitor would decrease his expenditures, in time or resources, in designing, producing or marketing a similar product.

d. The information consists of test data or other similar data concerning a process, method or component, the application of which results in a competitive advantage to FTI.

e. The information reveals special aspects of a process, method, component or the like, the exclusive use of which results in a competitive advantage to FTI.

f. The information contains ideas for which patent protection may be sought.

The document(s) listed on Exhibit "A", which is attached hereto and made a part hereof, has been evaluated in accordance with normal FTI procedures with respect to classification and has been found to contain information which falls within one or more of the criteria enumerated above. Exhibit "B", which is attached hereto and made a part hereof, specifically identifies the criteria applicable to the document(s) listed in Exhibit "A".

(iii) The document(s) listed in Exhibit "A", which has been made available to the United States Nuclear Regulatory Commission was made available in confidence with a request that the document(s) and the information contained therein be withheld from public disclosure.

(iv) The information is not available in the open literature and to the best of our knowledge is not known by Combustion Engineering, EXXON, General Electric, Westinghouse or other current or potential domestic or foreign competitors of FTI.

(v) Specific information with regard to whether public disclosure of the information is likely to cause harm to the competitive position of FTI, taking into account the value of the information to FTI; the amount of effort or money expended by FTI developing the information; and the ease or difficulty with which the information could be properly duplicated by others is given in Exhibit "B".

E. I have personally reviewed the document(s) listed on Exhibit "A" and have found that it is considered proprietary by FTI because it contains information which falls within one or more of the criteria enumerated in Paragraph D, and it is information which is customarily held in confidence and protected as proprietary information by FTI. This report comprises information utilized by FTI in its business which afford FTI an opportunity to obtain a competitive advantage over those who may wish to know or use the information contained in the document(s).

JOSEPH J. KELLY

State of Virginia)
) SS. Lynchburg
City of Lynchburg)

Joseph J. Kelly, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

JOSEPH J. KELLY

Subscribed and sworn before me
this 17th day of January 2000.

Brenda C. Cardona
Notary Public in and for the City
of Lynchburg, State of Virginia.

My Commission Expires July 31, 2003

EXHIBITS A & B

EXHIBIT A

Framatome Technologies, Inc. Topical Report BAW-10236P, "Addendum for Davis-Besse Repair Roll UTS Exclusion Zones," December 1999.

EXHIBIT B

The above listed document contains information, which is considered Proprietary in accordance with Criteria b, c and d of the attached affidavit.