February 20, 2002

Mr. Howard W. Bergendahl
Vice President-Nuclear, Davis-Besse
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1 - ISSUANCE OF AMENDMENT (TAC NO. MB2107)

Dear Mr. Bergendahl:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 252 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS). The amendment revises the technical specifications in response to your application dated May 22, 2001, and as supplemented by letters dated November 15, 2001, February 12, 2002, and electronic transmission dated February 19, 2002.

This amendment revises the DBNPS technical specifications in accordance with Framatome Technologies Incorporated Topical Report BAW-2303P, Revision 4, "OTSG Repair Roll Qualification Report." The requested changes would revise the existing DBNPS Once-Through Steam Generators (OTSGs) repair roll requirements to (1) use updated limiting tensile tube loads, (2) define new exclusion zones within the steam generator in which application of the repair roll is prohibited, (3) allow the repair roll to be used in the lower tubesheet area, (4) remove the limitation of only one repair roll per OTSG tube, and (5) replace the requirement that the repair roll be one inch in length with a requirement that the repair roll be installed in accordance with Topical Report BAW-2303P, Revision 4.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Stephen P. Sands, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures:
1. Amendment No. 252 to License No. NPF-3
2. Safety Evaluation

cc w/encls: See next page

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Docket No. 50-346

Enclosures: 1. Amendment No. 252 to License No. NPF-3
2. Safety Evaluation

cc w/encls: See next page

Distribution w/encls:

PUBLIC	GHill (2)	KManoly
PD3-2 r/f	WBeckner	FAkstulewicz
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ADAMS ACCESSION NUMBER: ML020450025

OFFICE	**PM:LPD3**	**LA:LPD3**	**SC:SRXB**	**SC:EMCB**	**SC:EMEB**
NAME	**S Sands**	**T Harris**	**FAkstulewicz***	**LLund***	**KManoly***
DATE	**02/20/02**	**02/20/02**	**02/ 15 /02**	**02/ 19 /02**	**02/ 19 /02**

OFFICE	**OGC**	**SC:LPD3**
NAME	**RWeisman***	**A Mendiola**
DATE	**02/ 19 /02**	**02/20/02**

OFFICIAL RECORD COPY

*See previous concurrence

cc:

Mary E. O'Reilly
FirstEnergy
76 South Main Street
Akron, OH 44308

Manager - Regulatory Affairs
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State - Route 2
Oak Harbor, OH 43449-9760

State of Ohio
Public Utilities Commission
180 East Broad Street
Columbus, OH 43266-0573

President, Board of County
Commissioners of Ottawa County
Port Clinton, OH 43216

Regional Administrator
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60523-4351

Michael A. Schoppman
Framatome ANP
1911 N. Ft. Myer Drive
Rosslyn, VA 22209

Resident Inspector
U.S. Nuclear Regulatory Commission
5503 North State Route 2
Oak Harbor, OH 43449-9760

Plant Manager, Randel J. Fast
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State - Route 2
Oak Harbor, OH 43449-9760

Dennis Clum
Radiological Assistance Section Supervisor
Bureau of Radiation Protection
Ohio Department of Health
P.O. Box 118
Columbus, OH 43266-0118

Carol O'Claire, Chief, Radiological Branch
Ohio Emergency Management Agency
2855 West Dublin Granville Road
Columbus, OH 43235-2206

Ohio Environmental Protection Agency
DERR--Compliance Unit
ATTN: Zack A. Clayton
P.O. Box 1049
Columbus, OH 43266-0149

Attorney General
Department of Attorney
30 East Broad Street
Columbus, OH 43216

Director
Ohio Department of Commerce
Division of Industrial Compliance
Bureau of Operations & Maintenance
6606 Tussing Road
P.O. Box 4009
Reynoldsburg, OH 43068-9009

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 252
License No. NPF-3

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the FirstEnergy Nuclear Operating Company (the licensee) dated May 22, 2001, and as supplemented by letters dated November 15, 2001, February 12, 2002, and electronic transmission dated February 19, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-3 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 252, are hereby incorporated in the license. FirstEnergy Nuclear Operating Company shall operate the facility in accordance with the Technical Specifications.

3. In addition, the license is amended to add the following License Condition:

2.C(7) <u>Steam Generator Tube Circumferential Crack Report</u>

Following each inservice inspection of steam generator tubes, the NRC shall be notified by FENOC of the following prior to returning the steam generators to service:

a. Indication of circumferential cracking inboard of the roll repair.

b. Indication of circumferential cracking in the original roll or heat affected zone adjacent to the tube-to-tubesheet seal weld if no reroll is present.

c. Determination of the best-estimate total leakage that would result from an analysis of the limiting LBLOCA based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet reroll repairs, and heat affected zones of seal welds as found during each inspection.

FENOC shall demonstrate by evaluation that the primary-to-secondary leakage following a LBLOCA, if any, as described in Appendix A to Topical Report BAW-2374, July 2000, continues to be acceptable, based on the as-found condition of the steam generators. For the purpose of this evaluation, acceptable means that a best estimate of the leakage expected in the event of a LBLOCA would not result in a significant increase of radionuclide release (e.g., in excess of 10 CFR Part 100 limits). This is required to demonstrate that adequate margin and defense-in-depth continue to be maintained. A written summary of this evaluation shall be provided to the NRC within three months following completion of the steam generator tube inservice inspection.

4. This license amendment is effective as of the date of issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachments:
1. Changes to the Operating License
2. Changes to the Technical Specifications

Date of Issuance: February 20, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 252

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Insert the following page into the Operating License. The revised page is identified by an amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
- - - - - -	6a

2.C(7) Steam Generator Tube Circumferential Crack Report

Following each inservice inspection of steam generator tubes, the NRC shall be notified by FENOC of the following prior to returning the steam generators to service:

a. Indication of circumferential cracking inboard of the roll repair.

b. Indication of circumferential cracking in the original roll or heat affected zone adjacent to the tube-to-tubesheet seal weld if no reroll is present.

c. Determination of the best-estimate total leakage that would result from an analysis of the limiting LBLOCA based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet reroll repairs, and heat affected zones of seal welds as found during each inspection.

FENOC shall demonstrate by evaluation that the primary-to-secondary leakage following a LBLOCA, if any, as described in Appendix A to Topical Report BAW-2374, July 2000, continues to be acceptable, based on the as-found condition of the steam generators. For the purpose of this evaluation, acceptable means that a best estimate of the leakage expected in the event of a LBLOCA would not result in a significant increase of radionuclide release (e.g., in excess of 10 CFR Part 100 limits). This is required to demonstrate that adequate margin and defense-in-depth continue to be maintained. A written summary of this evaluation shall be provided to the NRC within three months following completion of the steam generator tube inservice inspection.

ATTACHMENT TO LICENSE AMENDMENT NO. 252

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3/4 4-9	3/4 4-9
3/4 4-9a	3/4 4-9a
B 3/4 4-3	B 3/4 4-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 252 TO FACILITY OPERATING LICENSE NO. NPF-3

FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By application dated May 22, 2001, and supplemented by letters dated November 15, 2001, February 12, 2002, and electronic transmission dated February 19, 2002, FirstEnergy Nuclear Operating Company (FENOC, or the licensee) submitted a request for changes to the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS), technical specifications (TS), in accordance with the proprietary version of Framatome Technologies Incorporated Topical Report BAW-2303P, Revision 4, "OTSG Repair Roll Qualification Report." The requested changes would revise the existing DBNPS Once-Through Steam Generators (OTSGs) repair roll requirements to (1) use updated limiting tensile tube loads, (2) define new exclusion zones within the steam generator (SG) in which application of the repair roll is prohibited, (3) allow the repair roll to be used in the lower tubesheet area, (4) remove the limitation of only one repair roll per OTSG tube, and (5) replace the requirement that the repair roll be one inch in length with a requirement that the repair roll be installed in accordance with Topical Report BAW-2303P, Revision 4.

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

By License Amendment No. 220 dated April 14, 1998, the Nuclear Regulatory Commission (NRC) staff approved the initial use of repair rolls in the upper tubesheet, as analyzed in BAW-2303P, Revision 3, for DBNPS. However, re-analysis of the original reroll methodology became necessary due to identification of loss of coolant accidents (LOCAs) that were more limiting than the design basis accident previously evaluated in BAW-2303P, Revision 3. In addition, the main steamline break (MSLB) transient has been re-analyzed, resulting in a new set of design loads. In BAW-2303P, Revision 4, Framatome Technologies, Inc. (FTI) described its analyses performed at the request of the B&W Owners Group (BWOG). The report was provided in support of the license amendment request for DBNPS for repair rolls to be installed in both the upper and lower tubesheets, to address multiple repair rolls in a single tube, and to change exclusion zones in which application of the repair roll is prohibited. The analysis in BAW-2303P, Revision 4, demonstrates that it is acceptable for a tube that has been repaired with a roll to slip under faulted conditions (but such a tube is not projected to slip under normal operating conditions), which constitutes a change in design criteria compared to the original evaluation. Repair rolls that have been installed under BAW-2303P, Revision 3, remain acceptable based on the criteria contained in BAW-2303P, Revision 4. (See Section 3.1 of this safety evaluation.)

By letter dated July 7, 2000, the BWOG submitted Topical Report BAW-2374, dated July 2000, (Reference 2), "Justification for Not Including Postulated Breaks in Large-Bore Reactor Coolant System Piping in the Licensing Basis for Existing and Replacement Once-Through Steam Generators." This topical report provides the risk-informed bases for excluding the large break loss of coolant accident (LBLOCA) from some design considerations. By letter dated March 12, 2001, the BWOG submitted the Topical Report BAW-2374, Revision 1, (Reference 8), "Risk-Informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping," dated March 2001. This topical report was used as the risk-informed basis for the license amendment application submitted under DBNPS Serial Number 2705, dated May 22, 2001. The NRC staff has not completed its review of either BAW-2374, dated July 2000, or BAW-2374, Revision 1, dated March 2001.

By letter dated February 12, 2002, (Reference 9), the licensee stated that Topical Report BAW-2374, dated July 2000, provides the risk-informed basis for the proposed repair roll design specified in Topical Report BAW-2303P, Revision 4, OTSG Repair Roll Qualification Report, that was referenced in the license amendment application submitted under DBNPS letter Serial Number 2705. Accordingly, since Topical Report BAW-2374, Revision 1, remains under NRC staff review, Topical Report BAW-2374, dated July 2000 (from here on, referred to as "BAW-2374"), submitted on behalf of the DBNPS, provides the risk-informed basis for the design specified in Topical Report BAW-2303P, Revision 4, that, in turn, provides the technical basis for this license amendment application. Therefore, the staff's evaluation presented in the following is based on its review of the Topical Report BAW-2374.

By only evaluating the reroll repairs for MSLB and small break LOCA (SBLOCA) faulted conditions, Topical Report BAW-2303P, Revision 4, implicitly credits the results of Topical Report BAW-2374 in excluding the LBLOCA from some design considerations. BAW-2374 explains that rerolled tubes may slip in the tubesheet during some LBLOCA scenarios if there is degradation (such as circumferential cracking) in the tube, and the slip would prevent the tube seal weld from carrying the axial load that results from the event. By letter dated November 27, 2000, the BWOG provided additional information related to BAW-2374.

Although the staff has not approved BAW-2374, based on the risk-informed arguments presented in BAW-2374, the staff accepts that the reroll repairs at DBNPS may slip during a LBLOCA, resulting in an increase in leakage past the reroll. FENOC has agreed to a License Condition that will require FENOC to demonstrate that the expected leakage following a LBLOCA, if any, continues to be acceptable, based on the as-found condition of their OTSGs. Section 3.4 of this safety evaluation (SE) contains the staff's evaluation of the risk-informed arguments presented in BAW-2374.

A LOCA in the hot leg is a potential challenge to OTSG tubes within the tube sheets, in the tube support plate regions, and in the regions where the tubes are not supported. In the last two regions, tube response to a LOCA is not impacted by reroll repairs. Consequently, the staff is not addressing LOCA issues associated with response of tube sections that are outside the constraints of the tube sheets. Such LOCA issues are being addressed during the NRC staff's review of BAW-2374, dated July 2000, and BAW-2374, Revision 1, dated March 2001.

2.0 BACKGROUND

DBNPS has two OTSGs that were manufactured by Babcock and Wilcox. The OTSG tubes were fabricated from Inconel Alloy 600 material and were restrained by the roll expansion joints in the upper and lower tubesheets. A hardroll process was used to expand the original tube-to-tubesheet rolls into the tubesheet. The expansion is about one to two inches in axial length. Each 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.

General Design Criterion (GDC) 14 of Appendix A to 10 CFR Part 50 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. While DBNPS was not specifically designed to the GDC, it was designed to requirements similar to GDC 14 as described in Appendix 3D of the DBNPS Updated Safety Analysis Report (USAR). A significant portion of the reactor coolant pressure boundary is maintained by SG tubes that have experienced various levels of degradation. Draft NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," provides guidance for an acceptable method for establishing the limiting conditions of tube degradation. In addition, the TS require periodic inspections of SG tubes. The TS also require that those tubes with defects in excess of the repair limits (e.g., flaws equal to or greater than 40 percent through-wall) be repaired or removed from service.

The original joint between the tube and tubesheet is an interference fit constructed by roll expanding the tube into the bore of the tubesheet, followed by a seal weld at the primary face of the tubesheet. The undegraded original tube-to-tubesheet roll joint provides sufficient strength to maintain adequate structural and pressure boundary integrity.

Industry experience has shown that defects have developed in the tube-to-tubesheet roll joints as a result of various degradation processes. In general, tubes with degraded roll joints are either removed from service or repaired. The NRC has accepted alternate repair criteria allowing repaired tubes with degraded roll joints to remain in service provided that the repaired tubes can maintain adequate structural and leakage integrity under loadings from normal operation, anticipated operational occurrence, and postulated accident conditions. Such roll joints are said to be "qualified."

RG 1.121 recommends that the margin of safety against tube rupture under normal operating conditions be equal to or greater than three 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 (ASME). Structural loads imposed on the tube-to-tubesheet roll under normal operating conditions primarily result from the differential pressure between the primary and secondary sides of the tubes. Cyclic loadings from transients (e.g., startup/shutdown) were also considered in the qualification of the roll joints.

3.0 EVALUATION

3.1 Qualification Program

For the previous license amendments granted for rerolling at DBNPS, the licensee performed a qualification program presented in BAW-2303P, Revision 3, that demonstrated the strength of the roll joints was satisfactory in accordance with RG 1.121. Hydrostatic pressure tests were performed on mockup samples at a pressure value exceeding both 3 x normal operating pressure and 1.43 x MSLB pressure. No mechanical change or gross leakage in the samples was noted. Room temperature leak tests, thermal and fatigue cycling were conducted. The effect of tube hole dilation on joint strength was also evaluated and an exclusion zone was defined to ensure that the rerolled 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.

For the current license amendment request, FTI developed a qualification program presented in Revision 4 of BAW-2303P to demonstrate repair roll joint integrity through slip and leak tests. The program consisted of (1) establishing tube loads for the qualification tests, (2) preparing a mockup to simulate tubesheet conditions for qualification tests, and (3) performing verification tests and analyses.

The current license amendment request would revise the existing DBNPS OTSG repair roll requirements to (1) use updated limiting tensile tube loads, (2) define new exclusion zones within the SG in which application of the repair roll is prohibited, (3) allow the repair roll to be used in the lower tubesheet area, (4) remove the limitation of only one repair roll per OTSG tube, and (5) replace the requirement that the repair roll be one inch in length with a requirement that the repair roll be installed in accordance with Topical Report BAW-2303P, Revision 4.

FTI utilized the ANSYS finite element (FE) modeling code, and a linear-elastic, axisymmetric model of an overall OTSG, including the tube bundle, the tubesheets, shell, heads, and support skirt, to quantify the general structural behavior of the OTSG during various operating and accident transients. The ANSYS FE modeling code is a well-known, commercial code with which the NRC staff is familiar from licensee submittals on other topics. In BAW-2303P, Revision 4, FTI provided a general summary of assumptions in the development of the FE model, assessments of the parameters addressing the significant effects of different features among the OTSGs, and a general summary of the results of the thermal-hydraulic and structural analyses. The NRC staff reviewed the inputs and assumptions used by the licensee to model the SG and the LBLOCA event. The staff found the inputs and assumptions to be acceptable. The staff also reviewed the results of the licensee's analysis. Based on the staff's engineering judgment, the licensee's results appeared reasonable and consistent with previous bounding analysis of tube loads predicted as a result of the reroll process.

In the qualification program, the impact of tubesheet bowing on the roll joints was considered, due to the fact that the tubesheet bore diameter can change during certain operating conditions. The combined effects of primary-to-secondary pressure differential and thermal loads may cause the tubesheet to bow in one direction or the other, which can lead the tubesheet bore to dilate or shrink. When the tubesheet bore is dilated, the contact stress between the roll joint and the tubesheet would decrease and, thereby, reduce the pullout resistance of the roll joint. The resulting bowing effect can produce a dilation of the tubesheet bore in the region of the tube-to-tubesheet joint, which may reduce the load carrying capability of the rolled joint. Various dilations were included in the test block to evaluate this behavior.

Based on the temperature differential between tubes and the shell, and the pressure differential across the circular (flat plate) tubesheets during normal operating and various transient conditions, the FE analyses provide axial tube loads and the data required to calculate tube and tubesheet bore dilations. FTI stated that the axial tube loads calculated by the FE analyses supersede all previously calculated axial loads.

FE thermal analysis was performed to model the general structural behavior of the OTSG, including deflections and axial tube loads, and the local structural behavior (hole dilations). The key results of the FE analyses included (1) axial tube loads as a function of tubesheet radial position, (2) tube-to-tubesheet hole differential dilations as a function of tubesheet radial position, and (3) tube-to-tubesheet hole differential dilations as a function of depth into the tubesheet. Differential dilation is a term that is used to refer to the interface between the tube outside diameter (OD) and the tubesheet bore diameter, which allows a comparison of the relative interface of the joint for any transient condition. The limiting accident transient for load-carrying capacity of the repair roll is a function of differential dilations and axial tube load, which are used to determine plant-specific exclusion zones for repair roll. The staff finds that the assumptions made in the development of the FE model and the reported results of the structural analyses are reasonable for the transients and accidents addressed in BAW-2303P, Revision 4.

A mockup was constructed that consisted of perforated cruciform metal blocks, which allowed simulation of tubesheet bore dilations by applying a biaxial load to the block. Alloy 600 tube samples were inserted into the block that simulated the tube-to-tubesheet configuration in the field. The tubes were expanded into the tubesheet using an expanding tool that had the same critical dimensions as the tool used in the field. The repair roll design and installation for Revision 4 of BAW-2303P is the same as that described in BAW-2303P, Revision 3. To obtain conservative leakage results, the sample tubes were roll expanded using a spacer such that there was no heel transition in the tested repair roll. By removing the heel transition, the tested condition represented a complete circumferential severance at the end of the effective roll (primary side). After tube installation, the blocks were thermally cycled. The thermal cycles represent the effects of heat-up and cooldown cycles.

Testing was performed with a clean crevice between the OD of the tube samples and the tubesheet bore. Using a clean crevice was determined to be conservative, based on a summary of the results of a proprietary analysis conducted in 1999 using the same repair roll installation process as that currently used for the OTSGs. Because the licensee requested the removal of the restriction on lower tubesheet area rerolling, it presented the following results from tests performed to evaluate the effects of crevice deposits on leakage and joint strength.

Leak tests were performed for samples with and without crevice deposits, pre-fatigue and post-fatigue, using a representative material in the crevice deposits. The leak tests showed that for the OTSG repair roll installation process, a clean crevice leaks more than a packed crevice, both in the pre-fatigue and post-fatigue cases. The decreased leakage for the packed crevice is attributed to sludge providing a partial seal between the tube and tubesheet that would be an open flow path in a clean crevice. Similarly, the joint strength test results showed that the pre-fatigue, clean crevice sample resulted in the minimum joint strength. Based on 1999 test results provided by the licensee, the staff found that the test plans with the clean crevice in the post-fatigue case, as described in Revision 4 of BAW-2303P, would bound the leakage analysis and that lower tubesheet area rerolling is acceptable.

The FE analyses' results were reviewed to determine a bounding set of dilation test cases. Then a set of corresponding bounding axial loads were developed, which together with the tubesheet bore dilations effectively bound the normal operating and selected accident transients for the OTSG. The test matrix was developed from a set of applied loads for each slip test case and a combination of internal pressure and applied load for leak tests. The test sequence progressed from less severe conditions (tubesheet bore dilations and/or axial loads) to more severe conditions. When tube movement was noted, the initial sequence of tests was terminated for that sample. FTI performed testing to (1) measure the loads at which tube slippage would occur, (2) measure leakage for reroll joints that did not slip, and (3) measure leakage if tube slippage did occur. The test data were compiled and summarized to develop slip and leak criteria to qualify installation of a repair roll on a plant-specific basis. The repair roll is allowed to slip under specific faulted conditions.

To quantify leak rates for repair rolls subjected to accident conditions, applicable tubesheet bore dilations were achieved with representative pressures adjusted for uncertainties. The maximum pressure differential provides a bounding leak rate for all transients. The tube end was sealed so that the leak path was through the repair roll. To obtain conservative leakage results, the sample tubes were roll expanded using a spacer such that there was no heel transition in the tested repair roll. Leak tests were performed at room temperature. Room temperature leak tests are expected to be conservative based on higher temperatures increasing the joint tightness due to thermal expansion differences between the Inconel 600 tubes and the carbon steel tubesheet.

To verify that the repair roll could withstand anticipated axial loads during normal operation and accident conditions, applicable tubesheet bore dilations were achieved and an axial load was applied using a swage-lock fitting or an inside diameter gripper attached to the free end of the tube. A full circumferential severance was modeled for the testing, which is conservative for structural and leakage integrity since the majority of the degradation within the tubesheet is from short, axial cracks. The testing assesses the joint strength of a repair roll without taking any credit for the original roll expansion or the tube-to-tubesheet weld. Tube movement was monitored during the test and verified by measuring the depth of the tube end after each test.

On the basis of its qualification program, the licensee established that either a single or double roll repair will carry all structural loads and minimize potential leakage. For a double roll, a second repair roll is installed that overlaps a single repair roll. Both the single and double repair rolls may be installed in the upper tubesheet or lower tubesheet. The need to use a double roll depends on the location of the tube within the tube bundle. Using a double roll increases the joint strength because of the longer area of tube-to-tubesheet contact and the increased joint strength will accommodate larger applied loads. Having the option to use a double reroll in addition to the traditional use of a single reroll decreases the number of tubes that would be considered as part of an exclusion zone for applicability of reroll as an alternate repair criteria. The qualification program establishes bounding leak rates for rerolls longer than one inch, which the licensee will use in ensuring that it maintains leakage below TS limits, as further discussed below. Based on the qualification program results, the staff considers the elimination of the licensee's current requirement that the reroll be one inch in length to be acceptable.

In Reference 7, the licensee stated that DBNPS will evaluate each additional application of repair roll on a case-by-case basis as discussed in Section 9.0 item (f) and Section 2.1 of Topical Report BAW-2303P, Revision 4. The compressive loads due to the number of installed rolls in a tube when combined with the maximum compressive load at that tube location from the limiting transient (plant heatup) will not exceed the compressive loading design limit for the tubes with a margin of 50 lbs. DBNPS will manage compressive loads resulting from the repair roll process in accordance with Topical Report BAW-2303P, Revision 4. The root of concern for limiting compressive loads in tubes involves incorporating the compressive load imparted by plant processes, including low level heatup, which is identified as producing the most limiting compressive tube loads. Low-level heatup is not performed at DBNPS, and therefore there is additional compressive tube load margin, which changes the standard for maximum compressive load per tube. Based on this evaluation, the staff finds it acceptable to remove the limitation of only one reroll per SG tube from the existing DBNPS TS.

3.2 Structural and Leakage Integrity

Based on the results of the qualification testing, the licensee determined roll lengths sufficient to ensure adequate margins of structural and leakage integrity. The licensee determined the amount of slip for a tube with a new hardroll expansion based on the possible combination of loadings. Field experience to date has shown that the majority of the flaws in OTSGs within the tubesheet have been found to be short and axial in orientation.

With regard to structural integrity, the licensee demonstrated through slip tests that the limiting load for differential dilations consists of a major dilation and a minor dilation in the plane perpendicular to the tube. Differential dilations that are greater than the tested dilations were considered to be in an exclusion zone because test data were not available for such differential dilations.

With regard to leakage integrity, the qualification tests predicted a steady-state leak rate for each repair roll. The staff finds this approach acceptable because the predicted leak rate assumes a 360-degree, 100-percent through-wall circumferential flaw at the upper edge of the reroll and takes no credit for the original rolls or tube-to-tubesheet seal welds. Since most of the flaw indications in the original roll transitions have been found to be small and axially oriented, which are attributed to primary water stress corrosion cracking, the staff found that this is conservative and is an adequate approach. All tubes with an axial load in excess of the tested joint strength load are assumed to slip. In addition, a post-slip leak rate is applied without taking credit for the original roll or the tube-to-tubesheet weld.

A post-slip leak rate was applied to all repair rolls that have the potential to slip, regardless of whether a circumferential crack is actually present. The repair roll will not actually slip unless a large circumferential flaw is present. For the bounding load analyzed, the total leakage (and therefore the total number of repair rolls allowed) would be limited by the TS limits. The leak rate from each single repair roll or overlapping repair roll that is serving as a pressure boundary is summed to obtain a total leak rate for the OTSG.

By letter dated November 15, 2001, (Reference 7), the licensee described indications observed during previous outages relating to degradation in the tube roll joints in the upper and lower tubesheets in DBNPS SGs. During the DBNPS tenth refueling outage (10RFO) eddy current inspection in 1996, a single axial indication was detected in the upper roll transition of tube 58-119 in OTSG #2. The licensee determined that primary water stress corrosion cracking (PWSCC) caused the indication. The largest crack was 0.092 inches in length with a maximum depth of 78 percent through-wall (TW). Four other smaller axial PWSCC indications were observed which were below the eddy current testing (ECT) detection threshold ranging from 7 percent to 43 percent TW and 0.014 to 0.05 inches long. In addition, a shallow (3 percent TW) inner diameter intergranular attack (IGA) indication was observed in a circumferential band approximately 0.06 inches wide around the tube section examined.

Several metallurgical characteristics such as micro-hardness, residual stress, and cold work data from tube 58-119 were compared to that of rerolled mock-up specimens with and without post-roll stress relief heat treatments. It was learned that tube 58-119 was rolled into the tubesheet without a post-roll stress relief heat treatment. There is increased susceptibility to PWSCC relative to almost all other stress-relieved roll transitions at DBNPS, due to the lack of post-roll stress relief heat treatment. Fabrication records show that there are currently six other roll transitions at DBNPS that have been identified as being non-stress relieved.

It was identified in NRC Information Notice 98-27 that during the 11RFO in 1998, the licensee observed axial tube end internal diameter initiated cracking. Results from rotating coil eddy current inspections revealed the cracking to be in the heat affected zone of the tube to upper tubesheet weld. According to FirstEnergy, five tubes with these flaws were discovered in 1998. Four of the tubes with the tube end cracking were repaired with reroll and the fifth tube plugged due to a volumetric defect. An additional eight indications of this type damage were discovered in 2000 during 12RFO. All tubes with these indications, including those rerolled in 11RFO, were removed from service by plugging. Eddy current inspection was repeated for the four tubes that were rerolled in 11RFO during 12RFO, to look at tube end indications. The eddy current data from 12RFO was compared to that obtained during 11RFO and showed no discernable change in the flaws during one cycle of operation. The 12RFO inspection of the repair rolls installed in 11RFO did not identify any degradation following one cycle of service. To date, no circumferential crack-like indications have been identified in DBNPS upper tubesheet roll transitions or tube ends.

According to FirstEnergy, there have been no indications of degradation in the rolled joints in the lower tubesheet. Although rotating probe eddy-current inspections of the rolled joints have not been performed, an analysis has been prepared to determine when indications of degradation may appear. The analysis uses a predictive methodology for determining PWSCC initiations. The analysis is based on the time (duration, effective full power year (EFPY)) at which degradation was first observed in the upper tubesheet and the differences in temperature between the upper and lower tubesheets. The staff recognizes that there are large uncertainties in applying predictive methodologies to PWSCC initiation. This should be supported by a well defined inspection program to benchmark the prediction method at the plant to which it is applied.

All inservice repair rolls will continue to be inspected for degradation during each future inspection of the original OTSGs as presently required by TS Surveillance Requirement 4.4.5.9.

3.3 Field Installation and Inspection

The licensee proposed to repair degraded roll joints in the same manner as those repairs specified in Revision 4 of BAW-2303P. The licensee will install either one or two hardroll joints (reroll) in the tubes that have degradation in or near the original roll or reroll region. The repaired 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 torque is automatically controlled during the rerolling and is recalibrated after installation of a certain number of rerolls to ensure the minimum torque is maintained to produce proper fit.

After the installation, the licensee will inspect all rerolls using a bobbin coil probe and a plus point probe to ensure proper diametral expansion and positioning of the reroll repair joint. In addition, the inspections will verify that the reroll regions are free of degradation. For future inservice inspections, the licensee will inspect all rerolled tubes during SG inspection activities.

The rerolled region forms the new pressure boundary for the tube. As a result, conditions outboard the rerolled joint would not be expected to affect the integrity of the pressure boundary. However, conditions such as severe denting between the original roll and the rerolled region (or two rerolled regions) could be postulated to occur that may affect the load bearing capability of the joint. As a result, it remains important to monitor this region for conditions that could affect the integrity of the joint.

3.4 LBLOCA Considerations

In the above discussions of tubesheet hole dilations and leakage evaluations, the faulted conditions under consideration were limited to MSLB and SBLOCA. BAW-2303P, Revision 4, does not evaluate the performance of rerolls following a LBLOCA. Instead, BAW-2303P, Revision 4, implicitly credits Topical Report BAW-2374, which provides risk-informed arguments to justify excluding the LBLOCA from consideration as a faulted condition. The staff has not approved BAW-2374 for referencing in a plant's licensing basis. However, the staff has reviewed the risk-informed arguments in BAW-2374 as they relate to the reroll repairs at DBNPS. The staff performed its review in accordance with Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," as described below.

RG 1.174 contains general guidance for using probabilistic risk assessments (PRAs) in risk-informed decision-making on plant-specific changes to the licensing basis, and defines a licensing basis change as "modifications to a plant's design, operation, or other activities that require NRC approval." RG 1.174 also provides an acceptable approach to analyzing and evaluating proposed licensing basis changes.

According to the guidelines in RG 1.174, the applicant proposing the licensing basis change should perform an analysis of the proposed change using a combination of traditional engineering analyses with supporting insights from PRA methods. RG 1.174 states that when using risk-informed decision making, the proposed changes are expected to meet a set of key principles. These principles are:

a. The change meets current regulations unless an exemption is requested.

b. The change is consistent with the defense-in-depth philosophy.

c. The change maintains sufficient safety margins.

d. The increase in core damage frequency (CDF) or risk is small.

e. The impact should be monitored using performance measurement strategies.

3.4.1 Evaluation

BAW-2374, which in the context of this amendment is used to support the technical basis of the proposed reroll design specified in BAW-2303P, Revision 4, eliminates consideration of the thermal and pressure loads that result from a postulated LBLOCA. While loading conditions resulting from pipe break events are not included in design considerations specified by the ASME Code, they are required by NRC regulation and evaluated in accordance with ASME Code principles. Pressure loads are classified as primary stresses per the ASME Code and the Code requires evaluation of primary stresses for all conditions. However, the pressure loads on an OTSG are small during a LBLOCA when compared to other licensing basis events. Therefore, excluding the pressure loads resulting from a LBLOCA would not result in a decrease in the existing structural margins. However, due to differential thermal expansion during a LBLOCA event, significant thermal stresses may develop in some SG components. While ASME Code guidelines would classify these thermal stresses as secondary stresses and permit them to be excluded from the structural analysis when considering faulted conditions, the staff has taken the position that for SG tubes and tube repair methods, including rerolls, these thermally-induced stresses are significant and should be considered in facility licensing bases. Eliminating consideration of the thermal stresses resulting from a postulated LBLOCA from the design of SG tubes and tube repair methods could result in a decrease in design structural margins.

The staff has reviewed the engineering evaluations provided by the BWOG for the reroll repairs as described in Appendix D of BAW-2374. Although the information in BAW-2374 (particularly that regarding reroll operating experience) appeared to be applicable only to upper tubesheet reroll repairs, the BWOG confirmed by letter dated November 27, 2000, that all of the conclusions reached in the report were equally applicable to proposed lower tubesheet reroll repairs as well.

The engineering analysis in BAW-2374 regarding the performance of rerolls during the LBLOCA assumed that the tube was completely severed just to the primary system side of the reroll repair. The BWOG considered this to be a conservative assumption since, to date, no reroll repair has been installed in a tube with a complete severance. In addition, critical flaw size calculations have suggested that a very large circumferential flaw would have to exist (approximately 60 percent through-wall and 150 degrees in extent) for the LBLOCA loads to cause tube severance. If complete severance did not occur just to the primary system side of the reroll repair, additional margin beyond that discussed below would exist, since load could be transmitted to the original roll joint and fillet weld.

Considering only the structural integrity provided by the reroll repair joint, in the event of the limiting LBLOCA the axial (differential thermal expansion) loads and dilations placed on the reroll joint would result in the load carrying capacity of the joint being exceeded. As a result, the reroll joint would be expected to slip within the tubesheet until the displacement-controlled thermal expansion loads were relieved. This was conservatively estimated by the BWOG to entail a slippage of approximately 1.5 inches. Hence, the BWOG concluded that, provided current exclusion zone criteria in BAW-2303P are followed (which do not permit reroll repairs within 2 inches of the secondary-side face of the tubesheet), the reroll repair joint would remain within the tubesheet and the slipped tubes would not experience gross structural damage.

The BWOG also assessed the leakage integrity of the reroll repairs during the LBLOCA. Because of the differences in tubesheet bore on tube dilation resulting from the thermal loads associated with the LBLOCA, some loss of connection between the two was expected. In the event that tube severance had occurred just to the primary system side of the reroll repair joint and joint slippage had occurred, this would permit leakage between the primary and secondary sides of the SG. However, during a LBLOCA, only a small pressure differential would be expected to exist wherein the primary side was at a higher pressure than the secondary side. Assuming a conservative gap between the rolled tube and tubesheet bore of 0.001 inch and a representative pressure and temperature, the BWOG topical report noted that the limiting leakage rate was determined to be 0.06 gallons per minute per slipped tube. Since the likelihood of a slipped tube was considered to be small and the leakage associated with a slipped tube was also considered to be small, the BWOG concluded that the leakage integrity of the reroll repairs was acceptable for LBLOCA events.

The staff examined the engineering evaluation provided by the BWOG. The staff determined that sufficient information had been provided to conclude that adequate structural integrity of the tube-to-tubesheet reroll joints could be maintained and that only a limited amount of leakage may be expected to result from joint slippage during a LBLOCA scenario. The staff concluded that the information provided in BAW-2374 regarding this subject is applicable to DBNPS. The staff noted that the BWOG estimate for leakage per slipped tube (0.06 gpm) appeared to be conservative. However, the staff concluded that the number of tubes expected to slip in the event of a LBLOCA was indeterminate.

During long-term cooling following a LOCA, emergency core cooling system (ECCS) water is drawn from the containment emergency sump by the low-head ECCS pumps. A 10 CFR 50.46 concern arises if leakage due to slipped tubes can jeopardize ECCS pump operation by reducing the sump water level. The NRC staff judges that thousands of gallons would have to be lost before this would occur. It further judges that licensees will achieve decay heat removal cooling within a few hours of a LBLOCA. If, for example, 100 tubes were to break within the tube sheets following a LBLOCA, the leak rate would be less than 6 gpm. Clearly, more than several hundred tubes would have to break and slip before there were serious consequences to ECCS operation. Breakage of this number of tubes is not credible. The NRC staff finds that reroll repairs do not cause a 10 CFR 50.46 concern.

To date, reroll repairs have only been installed in upper tubesheets. Operational experience has shown that the majority of cracking at elevations above reroll joints has been axially-oriented cracking in the primary-side (upper) reroll transition region of the reroll and axially or circumferentially-oriented cracking in the heat affected zone (HAZ) of the tube-to-tubesheet fillet weld. The instances of circumferentially-oriented cracking in the fillet weld HAZ have been limited in extent and insufficient to lead to complete tube severance. However, this does not preclude the possibility that future in-service inspection results may show more significant circumferential cracking in the weld HAZ region, in the secondary-side (lower) transition region of original upper tubesheet rolls, or in either the primary-side (upper) or secondary-side (lower) reroll transition region of upper tubesheet reroll repairs. Similar conclusions can be made regarding the potential for degradation in lower tubesheet original roll and reroll regions as well.

Present SG inspection activities will be continued to ensure that, should significant circumferential cracking occur in the HAZ in the future, it will be identified, evaluated, and reported to the staff. Pending the completion of the review of BAW-2374, it is the staff's position that when individual licensees intend to install reroll repairs, a best-estimate evaluation must be performed to demonstrate that the as-found condition of the licensee's SGs (based on the most recent inspection results) is such that the technical bases for concluding that the amount of leakage (as described in Sections 3.4.1.1 and 3.4.1.2) that would occur in the event of a LBLOCA would be acceptable. DBNPS, through the acceptance of a License Condition associated with this safety evaluation, will perform evaluations based on the results of ongoing SG inspections to meet this staff position.

3.4.1.1 Defense-in-Depth Considerations

Based on current SG inspection, evaluation, and degradation management practices (e.g., plug-on-detection criteria for SG tubes exhibiting circumferential cracking), the NRC staff concludes that DBNPS currently meets 10 CFR Part 100 limits and defense-in-depth concerns with regard to the SG tube end cracking issues which the reroll repair option was designed to address. Therefore, barring the onset of unanticipated SG tube operational leakage, the staff concludes that DBNPS will meet 10 CFR Part 100 limits until such time as the unit shuts down and, consistent with this license amendment, may effect reroll repairs. This license amendment will permit the licensee to repair and leave in service at DBNPS, SG tubes with circumferentially oriented tube end cracking. Accordingly, as set forth below, the NRC staff will require via a License Condition, that the licensee specifically evaluate the as-found condition of their SGs based on future inspection results and loads resulting from the postulated LBLOCA to ensure that 10 CFR Part 100 limits will continue to be met.

BAW-2374 demonstrates that rerolls could slip and leak following a LBLOCA, but this would not result in a significant degradation of the SG tube pressure boundary. BAW-2374 also explains that traditional defense-in-depth considerations would be maintained, specifically that a sequence of independent failures must occur in order for core damage or large radiological release to result from tube damage during a LBLOCA. For core damage to result, these events include the extremely low frequency pipe rupture event itself, a secondary side isolation failure, and a failure of recovery actions that would prevent sump depletion (which would take considerable time for the leak rates discussed above). For large early release, the failures include the pipe rupture, a failure of secondary system isolation, a failure of the ECCS low pressure recirculation system, and an unscrubbed release pathway via the secondary side/balance of plant (note that the leakage past the reroll repairs is a tortuous path).

BAW-2374 also demonstrates that, when considering the spectrum of LBLOCAs, the limiting rupture size/location from the standpoint of causing rerolls to slip does not correspond to the limiting rupture size/location from the standpoint of potential core damage (e.g., limiting in

10 CFR 50.46 analysis). Hence, while containment integrity may be slightly diminished as a result of reroll joint slippage, the likelihood of extensive fuel cladding failure from the less challenging LBLOCA scenario is also diminished.

In addition, the proposed amendment includes a License Condition (see Section 3.5 of this safety evaluation (SE)) that requires the licensee to demonstrate that, based on the condition of the SGs as determined by inspection, the amount of leakage expected in the event of a LBLOCA at DBNPS, if any, will continue to be acceptable. In this context, “acceptable leakage” means that the best estimate leakage (in the event of a LBLOCA) would not result in a significant increase in radionuclide release (e.g., in excess of 10 CFR Part 100 limits). For these reasons, the staff finds that this evaluation will demonstrate that adequate safety margins and defense-in-depth will continue to be maintained in the design and installation of the reroll repairs at DBNPS.

3.4.1.2 Safety Margins

BAW-2374 noted that the design and repair of OTSGs will continue to be governed by the requirements of Section III and Section XI of the ASME Boiler and Pressure Vessel Code along with staff guidance provided in draft NRC RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." In general, draft RG 1.121 is based on the requirements of the ASME Code, and in addition, specifies that SG tubes shall have a margin to burst of 3.0 for normal operating conditions and a margin of 1.4 for faulted conditions.

Approximately 1 to 1.5 inches of reroll joint slippage would be expected if the original roll and fillet weld does not carry the axial loads. While permitting such slippage does not maintain the same margins of structural integrity as the original roll and fillet weld (which would not slip), the staff concludes that the margins maintained are sufficient to ensure that reroll repairs will not cause a gross failure of the SG tube containment boundary. The staff also concludes that the evaluation of the expected leakage behavior of the reroll joints was reasonable, particularly considering the small differential pressures during the event. Finally, the License Condition included in the proposed amendment (see Section 3.5 of this SE) includes the licensee demonstrating, based on the condition of its SGs, that an acceptable amount of leakage would be expected in the event of a LBLOCA. In this context, “acceptable leakage” means the best estimate leakage (in the event of a LBLOCA) would not result in a significant increase in radionuclide release (e.g., in excess of 10 CFR Part 100 limits). For these reasons, the staff finds that sufficient safety margins will be maintained at DBNPS for reroll repairs in the event of a LBLOCA.

3.4.1.3 Change in Risk

BAW-2374 contains a bounding risk analysis to estimate the potential risk contribution (i.e., change in risk) by assuming a loss of OTSG tube integrity due to tube loads induced by LBLOCA (LBLOCA-induced SG tube rupture (SGTR)). The risk analysis uses CDF and large early release frequency (LERF) as the metrics for comparison to the acceptance guidelines of RG 1.174. In the risk analysis, event sequences associated with the postulated LOCA-induced SGTR scenarios were quantified to estimate the potential increase in CDF and LERF. This risk analysis conservatively assumes that the LOCA-induced SGTR is a catastrophic failure of the SG tube pressure boundary. This is significantly more challenging from the standpoint of losing coolant inventory than the limited SG pressure boundary leakage from rerolls described above.

In Section 3.4 of BAW-2374, two LOCA-induced SGTR scenarios were identified for quantitative assessment to determine the change in risk. Both scenarios begin with a LOCA in the upper region of the reactor coolant system (RCS) hot leg (i.e., "candy cane"). The RCS is refilled by the low pressure injection (LPI) subsystem of the ECCS, which induces a SGTR in the broken RCS loop. In the first scenario, secondary side isolation failure and failure of operators to initiate makeup water to the reactor building (RB) sump leads to eventual depletion of sump inventory through the secondary side, which causes ECCS failure and late core damage but no large early release. In the second scenario, secondary side isolation failure occurs, coupled with an independent failure of ECCS recirculation after depletion of the borated water storage tank (BWST), which leads to early core damage and possible large early release. The staff determined that these two LOCA-induced SGTR scenarios adequately represented the sequences of events for a bounding risk analysis of possible loss of OTSG tube integrity due to a large-bore RCS pipe break.

One other possible scenario, such as core damage caused by boron dilution from the secondary side (Generic Issue 141 of NUREG-0933, "A Prioritization of Generic Safety Issues"), was reviewed and determined to be not applicable to this issue for several reasons. When the SG tubes experience the high tube-to-shell differential temperature following the upper hot leg break, the secondary side pressure would be lower than the primary pressure (based on the BWOG's thermal-hydraulic evaluation contained in Appendix A to BAW-2374). This condition would not result in significant tube failures or allow sufficient leakage to dilute the RCS from the injection of non-borated water. Furthermore, there would be guidance in emergency operating procedures (EOPs) to terminate feedwater flow to the OTSG.

The change in CDF and LERF for the two LOCA-induced SGTR sequences were estimated by quantifying the cutset combinations containing the LOCA frequency, OTSG tube failure, secondary side isolation failure, failure of operator recovery actions to isolate the faulted SG and replenish primary inventory (in CDF sequence), independent failure of LPI recirculation (in LERF sequence), and the conditional probability of large early release. The staff reviewed the probability assumptions for each basic event in the cutset equations for the two scenarios and determined that conservative probability estimates for all of the basic events were used in the quantitative risk analysis.

BAW-2374 uses an initiating event frequency of 8×10^{-7} per reactor-year, which is based on a 36-inch large pipe using the Beliczey-Schulz correlation to account for the frequency of through-wall cracks in piping based on historical experience data (NUREG/ CR-5750) and the conditional probability of any rupture given a through-wall crack. This analysis assumed one through-wall crack to have occurred in a 36-inch diameter pipe, which was taken as conservative since, according to the topical report, “no TW (through-wall) cracks have been experienced in pipes larger than 8 inches.” The staff does not accept this basis for establishing the estimated frequency for 36-inch pipe breaks of 8×10^{-7} per calendar year because the staff has not concluded that the Beliczey and Schultz correlation alone provides a sufficient basis for calculating this frequency.

Instead, the staff concluded that the expected frequency for 36-inch pipe rupture is less than 1×10^{-6} per reactor-year. This conclusion is based on consideration of leak-before-break (LBB) approvals granted for all BWOG facility main coolant loops in the mid-1980s. LBB evaluations have been accepted by the staff, per the provisions of 10 CFR Part 50, Appendix A, GDC 4 to, "demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." In the rulemaking that implemented this provision into GDC 4, an extremely low probability of piping system rupture was equated to a frequency of 1 x 10^{-6} per reactor year or less. This probability is related to the frequency of failure of any location within the piping system analyzed for LBB. Consequently, the probability of a piping system rupture at the limiting location, in the "candy cane" portion of the main coolant loop hot leg, would be expected to be significantly less than 1 x 10^{-6} per reactor year. Hence, for the purpose of this safety evaluation, the staff accepts the BWOG estimated frequency for 36-inch pipe breaks of 8 x 10^{-7} per calendar year.

It should be noted, however, that through-wall cracking was recently discovered in a 34-inch main coolant loop hot leg to reactor pressure vessel nozzle weld at the V. C. Summer (Westinghouse design) facility, which may call into question certain conclusions that have been made regarding the frequency of large-bore piping rupture. The NRC staff will evaluate the results of the V.C. Summer root cause analysis to determine if any generic conclusions can be drawn regarding the probability of large-bore piping rupture. If generic implications are found, the NRC staff may conclude that it is necessary to reevaluate the technical basis for establishing large-bore pipe rupture frequencies.

Based on this LOCA frequency estimate and conservative probability estimates for other events in the cutset equations, the change in CDF was estimated to be 8 x 10^{-10} per reactor-year and the change in LERF was estimated to be 4 x 10^{-11} per reactor-year. These quantitative risk changes are considered as very small risk increases according to the risk acceptance guidelines in RG 1.174 and are acceptable.

3.4.1.4 Compliance with Regulations

To determine if an exemption was necessary, the staff considered whether permitting rerolled tube joints to slip complies with the regulations. While DBNPS was not specifically designed to the GDC, it was designed to requirements similar to the GDC as described in Appendix 3D in the DBNPS USAR. With respect to GDC-14, the staff concluded that the RCS pressure boundary at DBNPS continues to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Therefore, the results of the topical report confirms that GDC-14 continues to be satisfied. With respect to GDC-30, the staff has concluded that the reroll repairs at DBNPS are still designed and fabricated to the highest practicable standards as previously approved (on April 10, 1998) in Amendment No. 190. Therefore, GDC-30 is satisfied.

The staff also considered whether compliance with the requirements of 10 CFR 50.46 and Appendix K would be an issue at DBNPS, since the licensee did not request an exemption from these requirements. In the letter dated November 27, 2000, the BWOG discussed continued compliance with 10 CFR 50.46, given that the hot leg “candy cane” break scenarios described in Appendix A to BAW-2374 could result in leakage past rerolls. The BWOG presented analyses, based on quantitative sensitivity studies performed with FTI’s approved large break and small break evaluation models, and supplemented by qualitative analyses, to show that the consequences (with respect to 10 CFR 50.46(b)) of hot leg “candy cane” breaks are bounded by those of the existing licensing basis cold leg guillotine beaks.

With respect to the impact of reroll repairs, the staff concludes that these quantitative and qualitative analyses adequately demonstrate that the subject LOCA scenarios (large and small break) do not represent new limiting scenarios for ECCS performance. The staff also concludes that the qualitative and quantitative analytical methods employed by FTI satisfy the model requirements of 10 CFR 50.46(a)(1)(i) and (ii) for the specific analyses that the staff reviewed. Based on this, the staff concludes that LOCA analyses provided in support of BAW-2374 meet the requirements of 10 CFR 50.46 for DBNPS, and are, therefore, acceptable.

Based on this analysis, the staff has concluded that use of Topical Report BAW-2303P, Revision 4, which does not consider LBLOCA loads in the design of the reroll repairs, does not require an exemption to the regulations.

The staff notes that the analyses described in BWOG's letter dated November 27, 2000, relies on operator action, as instructed by plant EOPs, to achieve and maintain long term core cooling per 10 CFR 50.46(b). FirstEnergy has performed a review to assure that its EOPs are consistent with the descriptions in BAW-2374 in regard to the key operator actions for mitigation of the accident sequence of concern. The key operator actions are the transfer of the ECCS suction from the BWST to the containment sump, and the isolation of the secondary system to minimize any primary-secondary leakage. DBNPS has also confirmed that its EOPs are consistent with those in the BWOG letter referenced above. The staff concludes that FirstEnergy has sufficiently resolved the staff's concerns related to compliance with 10 CFR 50.46 for a LBLOCA and SBLOCA for reroll repairs at DBNPS.

3.4.1.5 Integrated Decision Making

The staff had considered removal of LBLOCA loads from the reroll design, consistent with the proposed use of Topical Report BAW-2303P, Revision 4, with respect to the integrated decision making criteria in RG 1.174. The staff has concluded that this change is a practical solution for addressing the thermal loads caused by a LBLOCA and their impact on the design of the SG repair method. Upon implementation of this amendment, the licensee will be able to use practical and acceptable repair methods (e.g., rerolls) at DBNPS and avoid premature plugging of SG tubes. This benefit outweighs the change in CDF and LERF, which is considered very small by RG 1.174 criteria. Further, adequate margin and defense-in-depth are maintained and 10 CFR 50.46 is met for reroll repair at DBNPS.

3.4.1.6 Implementation and Monitoring

The licensee has not proposed any changes to the existing monitoring programs. Implementation of the proposed amendment will not result in any changes in plant operation, inspections, or design. Inspection and monitoring programs which impact this safety evaluation can be broken down into two areas: (1) those that apply to RCS piping and (2) those that apply to SG primary-to-secondary pressure boundary.

RCS piping will continue to be inspected in accordance with the inservice inspection program, as required by the TS. Additionally, the DBNPS TS contain RCS leakage limits and require plant shutdown if those limits are exceeded. Leakage from repair rolls will be accounted for to ensure post-accident primary-secondary leakage will not exceed that assumed in the safety analyses. Further, primary system leakage is included within the performance indicators of the NRC's Revised Oversight Program, which ensures that appropriate emphasis will be given to any unacceptable change in RCS leakage. For these reasons, it is concluded that the existing inspection and monitoring programs at DBNPS will ensure a low probability of degradation of the RCS piping that could lead to a LBLOCA.

The SG tubes will continue to be inspected and plugged or repaired as required by the DBNPS TS. In addition, DBNPS must have an adequate inspection program for repaired tubes (including plugs) to verify that the primary-to-secondary leakage following a LBLOCA is within acceptable limits in order to provide an adequate basis for evaluating compliance with the technical arguments in BAW-2374 that were relied upon by this safety evaluation. Finally, the DBNPS TS will continue to require SG leakage limits and plant shutdown if the limits are exceeded. For these reasons, the staff has concluded that DBNPS's inspection and monitoring programs ensure that the SG primary-to-secondary pressure boundary will be adequately maintained to support the conclusions of this safety evaluation.

3.4.1.7 Conformance to RG 1.174

RG 1.174 describes an acceptable method for assessing the nature and impact of licensing basis changes by a licensee when the licensee chooses to support these changes with risk information. RG 1.174 identifies a four-element approach for evaluating such changes, and these four elements are aimed at addressing the five elements of risk-informed regulation. Staff review has determined that the risk-informed arguments in BAW-2374 that the staff relied on for this SE are consistent with RG 1.174 as discussed below:

Element 1: Element 1 of the RG 1.174 approach recommends that the licensee define the proposed change.

BAW-2303P, Revision 4, describes the faulted conditions that were evaluated in the design of the reroll repairs for DBNPS. BAW-2303P, Revision 4, implicitly relies upon BAW-2374 to exclude LBLOCA and only consider MSLB and SBLOCA as the faulted conditions. However, the staff has determined that the aspects of LBLOCA necessary to comply with 10 CFR 50.46 have been addressed for reroll repairs. The staff finds Element 1 is satisfied.

Element 2: Element 2 provides for the performance of an engineering analysis.

Under this element, the licensee performs a qualitative and quantitative analyses, traditional engineering approaches, and techniques associated with the use of PRA findings. Further, this element recommends that the licensee satisfy the principles set forth in Section 2 of RG 1.174. This includes, for example, establishment of a reasonable balance between prevention, mitigation, and avoidance of over reliance on programmatic activities.

Appendix A of BAW-2374 describes the thermal-hydraulic analysis of a LBLOCA with respect to its effect on primary-to-secondary leakage if SG tube leakage occurs. Appendix D of BAW-2374 describes the impact of the thermal-hydraulic loads on the SG tubes and repair methods. Based on these evaluations, the LBLOCA will only have a minor impact on the integrity of the SG reroll repairs (e.g., result in minor leakage). Further, the reroll repairs are still designed to handle MSLB and LOCAs of attached piping (SBLOCAs). Therefore, the SG tubes continue to mitigate the effect of an accident without over reliance on programmatic activities. The staff finds that the analysis criteria of this element are satisfied.

RG 1.174 states that in implementing risk-informed decision making, plant changes are expected to meet a set of key principles. The following paragraphs summarize these principles and the staff findings related to these principles.

- Principle 1 states that the proposed change must meet current regulations unless it is explicitly related to a requested exemption or rule change. The staff has concluded that permitting rerolls to slip during a LBLOCA meets the current regulations without requiring an exemption pursuant to 10 CFR 50.12. Therefore, principle 1 is satisfied.

- Principle 2 states that the proposed change must be consistent with the defense-in-depth philosophy. The staff has concluded that this amendment is consistent with the defense-in-depth philosophy in that (a) any LBLOCA-induced SG tube leakage would not result in a significant reduction in the effectiveness of the SG tube containment barrier, and (b) a sequence of independent failures would need to occur in order for core damage or large early release to occur. Therefore, principle 2 is satisfied.

- Principle 3 states that the proposed change shall maintain sufficient safety margins. The staff has concluded that the change maintains sufficient safety margins to ensure that gross failure of the SG tube containment boundary function does not occur. Therefore, principle 3 is satisfied.

- Principle 4 states that when proposed changes result in an increase in CDF or LERF, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement. The staff concludes that the increases in CDF and LERF are considered very small. Therefore, principle 4 is satisfied.

- Principle 5 states that the impact of the proposed change should be monitored using performance measurement strategies. DBNPS has not proposed any changes to the existing monitoring programs, which are already sufficient to monitor the integrity of the RCS pressure boundary and the SG tubes. However, DBNPS has committed to demonstrate that, based on the condition of the SGs, an acceptable amount of leakage is expected in the event of a LBLOCA. Therefore, principle 5 is satisfied.

Element 3: Element 3 is the definition of the implementation and monitoring program.

The primary goal of this element is to ensure that no adverse safety degradation occurs because of the proposed change. The staff has determined that the existing monitoring programs are sufficient to monitor the integrity of the RCS and SG tubes. However, DBNPS will, in accordance with the regulatory commitments included in the letter dated November 15, 2001, verify that the expected primary-to-secondary leakage is acceptable based on the current condition of their SGs. These regulatory commitments have been incorporated as a License Condition. Therefore, Element 3 is satisfied.

Element 4: Element 4 is the submittal of the proposed change.

DBNPS submitted a request for the change by letter dated May 22, 2001, through reference to Topical Report BAW-2303P, Revision 4, which relies on the technical arguments in BAW-2374. The application was supplemented by letters dated November 15, 2001, and February 12, 2002. Therefore, Element 4 is satisfied.

3.4.2 Summary of LBLOCA Considerations

In summary, the staff concludes that a break in the large-bore RCS hot leg could lead to large axial loads on the SG tubes due to the temperature difference between the SG tubes and the SG shell. The result, based on technical bases in BAW-2374 as applied to the reroll repairs at DBNPS would be a minor degradation in the SG tube pressure boundary in a condition where the pressure difference across the tubes is small.

The staff finds it acceptable that slippage within the tube sheet may occur during the limiting LBLOCA if other factors (such as circumferential cracking) prevent the tube seal weld from carrying the axial load. FENOC has agreed to a License Condition that requires them to demonstrate, prior to startup, based on the condition of the SGs (including the number of SG tubes that are expected to slip), that the total primary-to-secondary leakage following a LBLOCA, if any, continues to be acceptable (i.e., adequate margin and defense-in-depth continues to be maintained). For the purpose of this evaluation, “acceptable leakage” means that the best estimate leakage (in the event of a LBLOCA) would not result in a significant increase in radionuclide release (e.g., in excess of 10 CFR Part 100 limits). As a result of the License Condition described in Section 3.5 below, the staff finds that adequate margin of safety and defense-in-depth will continue to be maintained, and that the increase in risk as measured by CDF and LERF is small.

Additionally, excluding LBLOCA from consideration in designing rerolls is not intended to affect DBNPS’s approved LOCA evaluation models or the analysis performed to demonstrate compliance with the requirements of 10 CFR 50.46. It is not intended to exclude any pipe break sizes or locations from the DBNPS LOCA analyses that were performed to demonstrate compliance with 10 CFR 50.46 or exclude evaluation of consequent SG tube degradation from consideration in those analyses.

Therefore, the staff finds that the LBLOCA, as described in Appendix A to BAW-2374, does not need to be considered further in the design of DBNPS’s reroll repairs due to the License Condition as described in Section 3.5.

3.5 License Condition

Based on the licensee’s electronic transmission dated February 19, 2002, the licensee has agreed to the following License Condition:

Steam Generator Tube Circumferential Crack Report

Following each inservice inspection of steam generator tubes, the NRC shall be notified by FENOC of the following prior to returning the steam generators to service:

a. Indication of circumferential cracking inboard of the roll repair.

b. Indication of circumferential cracking in the original roll or heat affected zone adjacent to the tube-to-tubesheet seal weld if no reroll is present.

c. Determination of the best-estimate total leakage that would result from an analysis of the limiting LBLOCA based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet reroll repairs, and heat affected zones of seal welds as found during each inspection.

FENOC shall demonstrate by evaluation that the primary-to-secondary leakage following a LBLOCA, if any, as described in Appendix A to Topical Report BAW-2374, July 2000, continues to be acceptable, based on the as-found condition of the steam generators. For the purpose of this evaluation, acceptable means that a best estimate of the leakage expected in the event of a LBLOCA would not result in a significant increase of radionuclide release (e.g., in excess of 10 CFR Part 100 limits). This is required to demonstrate that adequate margin and defense-in-depth continue to be maintained. A written summary of this evaluation shall be provided to the NRC within three months following completion of the steam generator tube inservice inspection.

This License Condition will ensure that the licensee will perform an evaluation adequate to demonstrate that given as-found conditions in an outage, gross structural failure and leakage of the reroll repair joints will not occur in the event of a LBLOCA pending the resolution of the identified issue during the review of BAW-2374. This evaluation will demonstrate that adequate safety margins and defense-in-depth continue to be maintained in the design and installation of the reroll repairs at the DBNPS. It is recognized that further NRC review of BAW-2374 may require it to modify the License Condition or otherwise involve additional actions to comply with final NRC conclusion on the topical report.

3.6 Proposed Technical Specification Change

TS SR 4.4.5.4.a.7, TS SR 4.4.5.4.a.9 and Bases 3/4.4.5 would be changed as follows:

TS SR 4.4.5.4.a.7:

This section is revised by changing the repair roll process from only being used once per SG tube using a 1 inch reroll length to a process that is used to repair tubes with defects in the upper and lower tubesheet areas as described in topical report, BAW-2303P, Revision 4.

TS SR 4.4.5.4.a.9:

This section is revised by specifying that the previously existing roll is located outboard of the new roll area of the tubesheet, as opposed to above the new roll area. The rerolled region of the tube that will need to be inspected is specified to be only the length of tube inboard of the new roll area in the tubesheet. The primary system pressure boundary of a tube will be changed after reroll repair is performed. Thus, the portion of the tube outboard of the new roll can be excluded from future inspection because it will be outside of the pressure boundary.

Bases 3/4.4.5:

This page is revised by adding a reference to the implementation of reroll repair in accordance with BAW-2303P, Revision 4, and a definition of the outboard portion of a reroll joint.

The staff finds these changes acceptable based on the staff evaluation of Topical Report BAW-2303P, Revision 4, and limited review of Topical Report BAW-2374 for its application to the proposed reroll activity at the DBNPS, as discussed in this safety evaluation.

4.0 SUMMARY

The licensee proposed to implement an alternate repair method using a hardroll expansion process to repair tubes having indications of tube degradation in the original roll or repair roll (or both) regions of the upper or lower tubesheets. The technical basis for the proposed reroll method is documented in topical report BAW-2303P, Revision 4.

The staff has determined that (1) the licensee's alternate repair criteria were established on the basis of the qualification tests that used specimens simulating the actual tube-to-tubesheet joint configuration of the SGs, (2) the loads for structural and leakage tests were specified and applied in accordance with RG 1.121, and (3) the proposed changes to the TS satisfy all regulatory requirements applicable to SG tube integrity.

On the basis of submitted information, the staff concludes that the proposed TS changes regarding reroll repair for degraded roll joints in the SGs at DBNPS are acceptable because the licensee has demonstrated through an acceptable qualification program that the reroll satisfies GDC 14 of Appendix A to 10 CFR Part 50 and RG 1.121.

It should be noted, however, as indicated in Section 3.4.1.3 of this safety evaluation, through-wall cracking observed in a main coolant loop nozzle weld at the V.C. Summer facility may call into question conclusions that were made regarding the frequency of large-bore piping rupture. The nuclear industry, through the EPRI Materials Reliability Project (MRP), is addressing the issue of Alloy 82/182 piping weld primary water stress corrosion cracking as part of an ongoing industry program. By letter dated April 27, 2001, the MRP submitted the report, "PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for US PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," dated April 2001. The evaluation provided by the MRP was sufficient for the NRC staff to conclude that, based on available information, there was no immediate, generic safety concern related to this issue which required NRC staff action. However, the NRC staff is still awaiting MRP submittal of the final report on this topic, which may include recommendations for additional piping weld examinations using enhanced nondestructive techniques. Results of any additional examinations could modify the NRC staff's conclusions and the NRC may conclude that it is necessary to reevaluate the technical basis for establishing large-bore pipe rupture frequencies.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (66 FR 41621). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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

Principal Contributors: Anne Passarelli, EMCB, NRR
John Tsao, EMCB, NRR
Matthew Mitchell, EMCB, NRR
Pei-Ying Chen, EMEB, NRR
Warren Lyon, SRXB, NRR

Date: February 20, 2002

8.0 REFERENCES

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

2. Framatome Technologies Inc. Topical Report, BAW-2374, "Justification for Not Including Postulated Breaks in Large-Bore Reactor Coolant System Piping in the Licensing Basis for Existing and Replacement Once-Through Steam Generators," July 2000.

3. Framatome Technologies Inc. Topical Report, BAW-2303P, Revision 4, "OTSG Repair Roll Qualification Report," August 2000. (Proprietary)

4. Letter from D. Firth (Framatome) to U.S. Nuclear Regulatory Commission, "Additional Information to Topical Report BAW-2374, "Justification for Not Including Postulated Breaks in Large-Bore Reactor Coolant System Piping in the Licensing Basis for Existing and Replacement Once-Through Steam Generators," November 27, 2000.

5. Amendment Nos. 318, 318, and 318 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55, for Oconee Nuclear Station, Units 1, 2, and 3, dated December 15, 2000.

6. Letter from Guy Campbell (FirstEnergy) to U.S. Nuclear Regulatory Commission, "License Amendment Application to Revise Technical Specification 3/4.4.5, Steam Generators, Regarding Steam Generator Tube Repair Roll Requirements (License Amendment Request 01-0004)," May 22, 2001.

7. Letter from Guy Campbell (FirstEnergy) to U.S. Nuclear Regulatory Commission, "Supplemental Information Regarding License Amendment Application to Revise Technical Specification (TS) 3/4.4.5 Steam Generator Tube Repair Roll Requirements (License Amendment Request No. 01-0004; TAC No. MB2107)," November 15, 2001.

8. Framatome Technology Inc. Topical Report BAW-2374, Revision 1, "Risk-Informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping," dated March 2001.

9. Letter from Howard W. Bergendahl (FirstEnergy) to U.S. Nuclear Regulatory Commission, "Supplemental Information Regarding License Amendment Application to Revise Technical Specification (TS) 3/4.4.5 Steam Generator Tube Repair Roll Requirements (License Amendment Request No. 01-0004; TAC No. MB2107," Serial Number 2764, dated February 12, 2002.