April 12, 2004

Mr. Garry L. Randolph
Vice President and Chief Nuclear Officer
Union Electric Company
Post Office Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: SECONDARY SHIELD WALL OPENING MODIFICATION (TAC NO. MB9879)

Dear Mr. Randolph:

The Commission has issued the enclosed Amendment No. 161 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1 (Callaway). The amendment changes the operating license in response to your application dated June 27, 2003 (ULNRC-04868), as supplemented by letters dated December 9, 2003 (ULNRC-04926), January 14 (ULNRC-04939) and April 5, 2004 (ULNRC-04974).

The amendment approves the application of leak-before-break (LBB) for the accumulator and residual heat removal (RHR) lines, and installation of an opening in the secondary shield wall in terms of the effects of the opening on occupational exposure. The amendment does not approve the application of LBB for the pressurizer surge line and the ASCE 4-86 load combination methodology. The shield wall opening is related to plant modifications that would facilitate maintenance on the replacement steam generators (SGs) to be installed in Refueling Outage (RO) 14 (Fall 2005).

The dynamic effects associated with large reactor coolant system branch line ruptures can be excluded using the proposed LBB methodology only for the accumulator and RHR lines. For the pressurizer surge line, the pressurizer nozzle safe end to pipe weld has an Alloy 82/182 weld that is susceptible to primary water stress corrosion cracking (PWSCC) and LBB is not considered applicable to this line at this time because of that weld. In your letter of April 5, 2004, you amended your request to apply LBB to the pressurizer surge line by withdrawing the request for applying LBB to this weld and requesting that LBB be applied to the remainder of the line which is downstream of the pipe whip restraints needed for a double-ended break of the line at that weld. The enclosed Notice of Partial Withdrawal of Application for Amendment to Facility Operating License has been forwarded to the Office of the Federal Register for publication.

In considering your amended request for applying LBB to the pressurizer surge line, the NRC staff still has concerns about approving LBB because it would be applied to only a portion of the entire line (i.e., only that portion downstream of the pipe whip restraints needed for the Alloy 82/182 weld). To address the NRC staff's concerns, you are requested to demonstrate that effective mitigative measures are in place, or will be implemented. With such measures that are acceptable to the NRC staff, the NRC will reconsider its concerns about applying LBB to this line. You are requested to provide this information within 60 days of receipt of this letter. Because your staff has requested that the NRC staff approve the application of LBB for the accumulator and RHR lines as soon as practical for the April 2004 refueling outage, this partial approval of your application dated June 27, 2003, as amended by the letter dated April 5, 2004, is being issued.

The use of the American Society of Civil Engineers (ASCE) 4-86 "100-40-40" load combination methodology of combining components of seismic response loads is also not being approved at this time. In the conference call on April 1, 2004, the NRC staff identified to your staff the additional information on the proposed methodology that is needed to complete our review of the methodology.

In your application, it is stated that with the proposed LBB methodology (which included LBB for the pressurizer surge line), all concrete and reinforcing steel stresses, including reinforcing steel development length in the secondary shield wall plus opening, are within the applicable ACI 318-71 acceptance criteria for the existing steam generator loads. Since this amendment does not approve LBB for the pressurizer surge line, the documentation for the modification to the secondary shield wall must take into account that LBB for the pressurizer surge line has not been approved.

The amendment authorizes (1) the above change to the Callaway licensing basis, and (2) revisions to the Final Safety Analysis Report to reflect this change. There are no changes to the Callaway Technical Specifications.

In its letter dated November 3, 2003, to Hank A. Sepp, of Westinghouse Electric Company, the NRC stated that it would, in accordance with 10 CFR 2.790(b)(5), withhold the information designated as proprietary in the following topical reports (TRs), on eliminating the rupture of the pressurizer surge line, accumulator lines, and residual heat removal lines from the structural design basis for Callaway: WCAP-15983-P; Revision 0, WCAP-16019-P, Revision 0; and WCAP-16020-P, Revision 0; respectively. These TRs were submitted to the NRC in your application dated June 27, 2003.

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

Sincerely,

/RA/

Jack Donohew, Senior Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures: 1. Amendment No. 161 to NPF-30
2. Safety Evaluation
3. Notice of Partial Withdrawal

cc w/encls: See next page

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/RA/
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* See EMCB memo dated 02/13/2004
** See EMEB memo dated 03/12/2004
*** See previous concurrence

ACCESSION NO.: ML041050677 **NRR-058** (concurrence comments corrected)

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UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
License No. NPF-30

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

   A. The application for amendment by Union Electric Company (UE, the licensee) dated June 27, 2003, as supplemented by letters dated December 9, 2003, January 14 and April 5, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's 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, by Amendment No. 161, the license is amended to authorize revision of the Final Safety Analysis Report (FSAR), as set forth in the application for amendment by Union Electric Company dated June 27, 2003 and the supplements dated December 9, 2003, and January 14 and April 5, 2004. Union Electric Company shall update the FSAR to reflect the application of leak-before-break for the 10-inch accumulator and 12-inch residual heat removal lines, as described in the licensee's letters dated June 27 and December 9, 2003, and January 14 and April 5, 2004, and the NRC staff's safety evaluation dated April 12, 2004, in accordance with 10 CFR 50.71(e).

3. This amendment is effective as of its date of issuance and shall be implemented prior to entering Mode 4 during the startup from Refueling Outage 13 which is scheduled for the Spring of 2004.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Robert Gramm for/

Stephen Dembek, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: April 12, 2004

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By application dated June 27, 2003, as supplemented by letters dated December 9, 2003, January 14 and April 5, 2004 [References 1 through 4], Union Electric Company (the licensee) requested changes to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1 (Callaway). The licensee is proposing to amend the operating license to allow plant modifications in order to facilitate maintenance on the replacement steam generators (SGs) to be installed in Refueling Outage (RO) 14 (Fall 2005). The proposed modifications would (1) replace the existing sludge lance platforms with new platforms to provide a larger platform area around each SG, and (2) cut a permanent access opening through the secondary shield wall to improve access to the sludge lance platforms. These modifications are to be done in RO 13 (Spring 2004).

To support the plant modification of the secondary shield wall, the licensee has also requested approval of the following changes to the Callaway licensing basis:

- Use of leak-before-break (LBB) methodology pursuant to General Design Criterion (GDC) 4 (of Appendix A to 10 CFR Part 50) so that dynamic effects associated with several large reactor coolant system (RCS) branch line ruptures may be excluded.

- Use of American Society of Civil Engineers (ASCE) 4-86 "100-40-40" method for combining the seismic response load components in place of the current square-root-sum-of-the-squares (SRSS) method.

The amendment would be the authorization of changes to the Callaway licensing basis (see Attachment 3 to the application) to be added to the Callaway Final Safety Analysis Report (FSAR). There are no proposed changes to the technical specifications (TS).

The licensee's description of the proposed changes, technical analysis, and regulatory analysis in support of its proposed license amendment are given in Sections 2.0, 4.0 and 5.2, respectively, of the licensee's application. The detailed evaluation below will support the conclusion 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public. The additional information provided in the supplemental letters dated December 9, 2003, and January 14 and April 5, 2004, do not expand the scope of the application as noticed and do not change the NRC staff's original proposed no significant hazards consideration determination published in the *Federal Register* on July 22, 2003 (68 FR 43397).

In its letter dated November 3, 2003, to Hank A. Sepp, of Westinghouse Electric Company, the NRC stated that it would, in accordance with 10 CFR 2.790(b)(5), withhold the information designated as proprietary in the following topical reports (TRs), which were submitted to the NRC in the licensee's application dated June 27, 2003 [References 5, 6, and 7]:

- WCAP-15983-P, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," Revision 0.

- WCAP-16019-P, "Technical Justification for Eliminating 10" Accumulator Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant," Revision 0.

- WCAP-16020-P, "Technical Justification for Eliminating 12" Residual Heat Removal (RHR) Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant," Revision 0.

2.0 REGULATORY EVALUATION

The proposed amendment is for the NRC staff to consider changes to the licensing basis for Callaway for an opening in the secondary shield wall, the application of LBB for certain large RCS branch line ruptures, and the application of ASCE 4-86 "100-40-40" seismic load combinations. Therefore, the regulatory requirements, which are discussed below, are given in terms of the following: (1) the occupational exposure from an opening added to the secondary shield wall, which is used to reduce occupational exposure doses inside containment outside the wall, (2) the application of LBB methodology to several large RCS branch lines, and (3) the application of ASCE 4-86 "100-40-40" seismic load combination methodology.

Occupational Exposure

Significant modification to plant radiation shielding design presents two regulatory issues. The first issue is whether the revised plant design can support the conclusion that the design features of the nuclear power plant is such that radiation doses to individuals can be maintained within the limits in 10 CFR 20.1201 and that the licensee makes every reasonable effort to maintain radiation exposure as low as is reasonably achievable (ALARA), as required in 10 CFR 20.1003, during normal operations and anticipated operational occurrences (AOOs). The second regulatory issue is whether the final radiation shielding is sufficient to allow personnel access to vital areas during accident conditions. The acceptance criteria is from the lessons learned from the Three Mile Island (TMI) accident in NUREG-0737, Action Item II.B.2. These regulatory requirements are discussed in Sections 12.1, "Assuring That Occupational Radiation Exposures Are as Low as Is Reasonably Achievable," and 12.3-12.4, "Radiation Protection Design Features," of the NRC Standard Review Plan (SRP) in NUREG-0800. There are other regulatory requirements given in SRP Sections 12.1 and 12.3-12.4, but they are not applicable to this proposed amendment.

LBB Methodology

GDC 4 of Appendix A, "General Design Criteria for Nuclear Power Plants," of Title 10 to the Code of Federal Regulations Part 50 (10 CFR Part 50), states, in part:

> [h]owever, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Nuclear power plant licensees have, in general, been required to consider the dynamic effects which could result from the rupture of sections of high energy piping (i.e., fluid systems that during normal plant operations are at a maximum operating temperature in excess of 200°F and/or a maximum operating pressure in excess of 275 psig). This requirement has been formally included in GDC 4 which states, "[s]tructures, systems, and components important to safety....shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit."

The NRC modified GDC 4 to permit the dynamic effects of some high energy piping ruptures to be excluded from facility licensing bases based upon the demonstration of a extremely low probability of piping system rupture. Consistent with this modification to GDC 4, the NRC accepted the LBB analysis methodology as an acceptable means by which this extremely low probability of piping system rupture could be demonstrated. The philosophy of LBB behavior for high energy piping systems was developed by the NRC in the early 1980s, used in certain evaluations stemming from Unresolved Safety Issue A-2, "Asymmetric Blowdown Loads on PWR Primary Systems," and then subsequently expanded for application toward resolving issues regarding defined dynamic effects from high energy piping system ruptures. The methodology developed by the NRC for performing LBB analyses was detailed in NUREG-1061, Volume 3, which was published in November 1984 [Reference 9].

The criteria for the LBB analysis methodology being an acceptable means by which this extremely low probability of piping system rupture could be demonstrated is given in draft SRP 3.6.3, "Leak Before Break Evaluation Procedures" [Reference 8]. It is the criteria from NUREG-1061, Volume 3, which are listed in Section 3.2.2 of this safety evaluation (SE). The draft SRP also contains acceptance values for this criteria.

3.0 TECHNICAL EVALUATION

As stated in its application, the licensee is proposing the following changes to the licensing basis for Callaway:

1. A design modification to install a permanent access opening in the secondary shield wall to provide access to the sludge lance platforms around each SG in terms of occupational exposure.

2. Exclude the dynamic effects associated with a rupture of the pressurizer surge line, 10-inch accumulator lines, and 12-inch RHR lines from the structural design basis for Callaway using LBB methodology pursuant to GDC 4.

3. Use of the ASCE 4-86 "100-40-40" method for combining the components of seismic response loads, discussed in Draft Regulatory Guide DG-1108, "Combining Modal Responses and Spatial Components in Seismic Response Analysis" [References 14 and 15].

The amendment does not change the Callaway TSs. The amendment would authorize the licensee to include in the FSAR the descriptions of the secondary shield wall opening modification and the application of LBB to exclude the dynamic effects of certain large RCS branch line ruptures not currently approved for LBB. With the approval of these changes to the FSAR through approving the proposed amendment, the licensing basis for Callaway will be changed.

The permanent access opening in the secondary shield wall is to facilitate the maintenance of the replacement SGs to be installed in RO 14. The licensee would also replace the existing sludge lance platforms around each SG with new platforms to provide a larger platform area. The opening would be installed in RO 13. The licensee also requested approval for the use of LBB for certain large RCS branch lines and of the ASCE 4-86 "100-40-40" seismic load combination methodology because these changes are needed to support the shield wall opening modification.

For the ASCE 4-86 "100-40-40" seismic load combination methodology, the NRC staff has not completed its review of the methodology. This was discussed with the licensee in a conference call on April 1, 2004, in which the licensee agreed to the NRC staff completing its review after the upcoming refueling outage in April 2004. Although the licensee still wants this methodology to be approved for use at Callaway, the licensee stated that it does not need to have the methodology approved to do the modification in the upcoming refueling outage.

3.1 Design Modification

The licensee has proposed a modification to the containment which consists of creating a permanent opening in the secondary shield wall near the ‘C’ SG, replacement of the sludge lance platforms around each SG and the installation of a new walkway and ladder to provide access to the new larger platforms. This arrangement will allow personnel direct access to SG cubicles and provide more efficient worker traffic patterns, to facilitate SG maintenance. The licensee has requested approval for (1) the permanent opening in the secondary shield wall and (2) the analysis techniques used to evaluate this change to the plant. Below is the NRC staff's evaluation of the licensee's requests.

3.1.1 <u>Normal Operations and AOOs</u>

As discussed by the licensee's application and its supplements, the primary radiological consideration for cutting a personnel access opening in the secondary shield wall is the increased streaming of neutron and N-16 gamma radiation into spaces outside the ‘C’ SG cubicle during power operations. A locked cage is provided in the design to control access to the SG cubicles. Because shielding material is incorporated into the walls and floor of the access cage, to reduce the streaming through the access opening, the licensee has estimated that dose rates will increase in limited areas to 67 mrem/hour (at the 2026 foot elevation) and 188 mrem/hour (at the 2000 foot elevation). Both of these areas are inside containment, which is not normally occupied during power operations. In addition, access to containment will continue to be controlled as a high radiation area as defined in 10 CFR Part 20. During shutdown operations, the neutron and N-16 radiation is no longer present; therefore, the increase in dose rates outside the secondary shielding is not significant. Therefore, because of this and because the licensee also has a radiation protection program for Callaway designed to meet the occupational exposure limits in 10 CFR 20.1201, the NRC staff concludes that the proposed modification will not impair the licensee’s ability to meet the occupational exposure limits in 10 CFR 20.1201.

3.1.2 Post Accident Vital Area Access

The licensee has verified that no credit was taken for the secondary shield wall for dose calculations made to demonstrate post-accident vital area access in accordance with the requirements in NUREG-0737, Action Item II.B.2. Based on this, if the licensee does remove some shielding in the secondary shield wall to provide easier access to the SG cubicles, as proposed in the amendment, there will be no impact on the licensee’s vital area access analysis during accidents. Therefore, based on there being no change to the licensee's vital area access analysis, the NRC staff concludes that the Callaway design continues to meet TMI Action Item II.B.2.

3.1.3 As Low As is Reasonably Achievable

ALARA considerations were addressed in the licensee's application and the supplemental letter dated January 14, 2004. The proposed modification of installing an opening in the secondary shield wall and providing access to the sludge lance platforms will increase the accessibility spaces inside the secondary shielding walls, and facilitate SG maintenance. It will also eliminate the current practice of erecting temporary scaffold ladders each refueling outage to access sludge lance platforms. The more efficient, direct access to the SG cubicles and the shielded labyrinth entrance, are ALARA design features consistent with the guidance in Regulatory Guide 8.8, "Information Relevant To Ensuring That Occupational Radiation Exposures At Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

The licensee has estimated a 580 person-mrem overall collective dose savings, per refueling outage, associated with this plant modification. The bulk of this dose savings (500 person-mrem) is due to the shorter transit times associated with the improved access and traffic routing. The remainder of the dose savings is associated with the reduced need to erect scaffolding to support sludge lance operations. This dose savings compares favorably against the higher collective dose that may result from the increased dose rates in spaces outside the SG cubicles. The licensee considers the proposed design modification as an improved ALARA design feature of the facility.

Based on the above discussion, the NRC staff concludes that the proposed design modification meets the requirements on maintaining radiation exposures ALARA for occupational exposure in 10 CFR 20.1003.

3.1.4 Design Modification Conclusion

Based on the discussion above and its review of the proposed modification, the NRC staff concludes that the proposed change to the plant's radiation shielding design (1) will not impact the licensee's ability to access vital areas to mitigate the consequences of a design basis accident in accordance with NUREG 0737, Action Item II.B.2, (2) will not impair the licensee's ability to maintain the radiation doses to individuals within the limits in 10 CFR 20.1201, and (3) will allow the licensee to maintain radiation exposures ALARA for occupational exposure in accordance with 10 CFR 20.1003. Based on this, the NRC staff further concludes that the design modification of the secondary shield wall is acceptable.

3.2 LBB Methodology

As stated in GDC 4:

> [h]owever, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

For the purpose of this demonstration, the licensee submitted a LBB analysis prepared by Westinghouse for the 14-inch pressurizer surge line, 10-inch accumulator lines, and 12-inch RHR lines. LBB evaluations developed using the methodology contained in NUREG-1061, Volume 3, have been previously approved by the NRC as the demonstration of an extremely low probability of piping system rupture.

3.2.1 Identification of Analyzed Piping and Piping Material Properties

The following discussion contains information supplied by the licensee on its LBB analysis in its application and the attachments to the application. These attachments included the following proprietary reports prepared by Westinghouse for the Callaway plant: WCAP-15983-P, Revision 0; WCAP-16019-P, Revision 0; and WCAP-16020-P, Revision 0.

The licensee analyzed the following sections of piping for LBB behavior verification:

- The pressurizer surge line from its connection to the RCS hot leg to the pressurizer as shown in Figure 3-1 of WCAP-15983-P.

- The 10-inch accumulator lines from its connection to the RCS cold legs to the accumulator as shown in Figures 3-1 through 3-4 of WCAP-160190-P.

- The 12-inch RHR lines layout from its connections to the RCS hot leg line loops 1 and 4 as shown in Figures 3-1 and 3-2 of WCAP-16020-P.

The following information on the above pressurizer surge line, accumulator lines, and RHR lines was provided by the licensee in its application and responses to the NRC's request for additional information (RAI) on the application of LBB for these lines.

Pressurizer Surge Line

The pressurizer surge line piping was identified as having the following material components. The piping and fittings of the pressurizer surge lines were manufactured from wrought American Society of Mechanical Engineers (ASME) specification SA 376 Type 316 and SA 403 Type 304 stainless steel (SS). The welds in this system were identified as having been fabricated from SS with one inconel 82/182 weld at the nozzle safe-end using gas tungsten arc welding (GTAW)/shielded metal arc welding (SMAW) combination and GTAW/submerged arc welding (SAW) combination processes. The line was manufactured from 14-inch outside diameter, 1.251-inch nominal wall thickness pipe.

For the material properties used in the pressurizer surge line LBB evaluation, Westinghouse used minimum and average room temperature tensile properties based on Certified Materials Test Report (CMTR) data. The minimum and average tensile properties at temperatures of interest (205°F, 455°F, 617°F, and 653°F) were calculated using the ratio of the ASME Code Section II properties at room temperature to the Code properties at the temperatures of interest to scale the CMTR-based data. The modulus of elasticity variation with temperature was established based on ASME Code Section II values. The minimum tensile properties were used in the LBB critical flaw size determination, while the average tensile properties were used in the LBB leakage flaw size determination.

Accumulator Lines

The accumulator line piping was identified as having the following material components. The piping and fittings of the accumulator lines were manufactured from wrought ASME specification SA 376-Type 304, SA 312-Type 304, SA358 Type 304 and SA 403 Type 304 SS. The welds in this system were identified as having been fabricated from SS using GTAW and SMAW processes. The line was manufactured from 10-inch nominal diameter pipe, Schedule 140 with a minimum wall thickness of 0.896 inches.

For the material properties used in the accumulator line LBB evaluation, Westinghouse used minimum and average room temperature tensile properties based on CMTR data. The minimum and average tensile properties at temperatures of interest (70°F and 558°F) were calculated using the ratio of the ASME Code Section II properties and those tabulated in Table 3-1 of WCAP-16019-P. The modulus of elasticity values were established at various temperatures from the ASME Code Section II. The minimum tensile properties were used in the LBB critical flaw size determination, while the average tensile properties were used in the LBB leakage flaw size determination.

RHR Lines

The piping and fittings of the RHR line piping were manufactured from wrought ASME specification SA 376/SA 312 Type 304 and SA 403 Type 304 SS. The welds in this system were identified as having been fabricated from SS using GTAW and SMAW processes. The line was manufactured from 12-inch nominal diameter pipe, Schedule 140 with a minimum wall thickness of 1.005 inches.

For the material properties used in the RHR line LBB evaluations, Westinghouse used minimum and average room temperature tensile properties based on CMTR data. The minimum and average tensile properties at temperatures of interest (70°F and 619°F) were calculated using the ratio of the ASME Code Section II properties and those tabulated in Table 3-1 of WCAP-16020-P. The modulus of elasticity values were established at various temperatures from the ASME Code Section II. The minimum tensile properties were used in the LBB critical flaw size determination, while the average tensile properties were used in the LBB leakage flaw size determination.

3.2.2 Evaluation of LBB Factors

The evaluation of the LBB factors for the above three piping systems is based on the guidance and criteria in draft SRP 3.6.3 and NUREG-1061, Volume 3. The information on the above three piping systems is in the three Westinghouse WCAPs in References 5, 6, and 7, and in the licensee's response to the NRC staff's RAI dated December 9, 2003. The evaluation is broken down into the following two parts: (1) the mechanical engineering evaluation of snubbers, leak rate calculations, susceptibility to water hammer, low-cycle and high-cycle fatigue, and internal loads and locations, and (2) the materials engineering evaluation.

3.2.2.1 Mechanical Engineering Evaluation

As-Built Configuration

In accordance with draft SRP 3.6.3, Section III.1, the licensee verified in the supplementary letter dated December 9, 2003, that the LBB evaluations were based on the as-built dimensions of the piping systems, and that the pipe wall thicknesses satisfy the minimum ASME Code Section III wall thickness requirements. The NRC staff finds that this is acceptable.

Reliability of Snubbers

Draft SRP 3.6.3, Section III.1, requires a demonstration that snubber failure rates are maintained at a low level. In the supplementary letter dated December 9, 2003, the licensee discussed compliance with the snubber surveillance requirements at Callaway. The licensee stated that the Callaway snubber surveillance program is controlled by criteria stated in FSAR Section 16.7.2.1.1, and representative snubber samples are tested in accordance with these criteria, to ensure that failure rates are acceptably low. The three piping systems contain sixteen snubbers. Six of these snubbers have been functionally tested since initial plant operation and were found acceptable. Six other snubbers have been hand-stroked and these have passed, too. Two more snubbers are scheduled for functional testing during the next refueling outage. Based on this information, the NRC staff concludes that the licensee has demonstrated that compliance with the snubber surveillance requirements in the Callaway FSAR will provide assurance that snubber failure rates are and will be maintained at a low level.

Leak Rate Calculations

The leak rate calculations from cracks were determined based on methodology previously reviewed and approved by the NRC staff in Generic Letter (GL) 84-04 [Reference 10]. In that GL, the NRC staff stated that although leak rate calculations, especially for small cracks, are subject to uncertainties, the leak rate calculation scheme provides leak rates sufficiently large as to have a high probability of detection during normal operation. The leak rate calculation scheme was correlated with previously generated laboratory data, and compared with service data from leakage previously detected in pressurized water reactor (PWR) feedlines and a boiling water reactor (BWR) recirculation line. In addition, the licensee also stated that Callaway has a RCS pressure boundary leak detection system which is consistent with the guidelines of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," for detecting leakage of 1.0 gpm in one hour. This conforms with the requirements of draft SRP 3.6.3, Section III.3.

Susceptibility to Water Hammer

Draft SRP 3.6.3, Section III.4, requires an assessment of the susceptibility of the piping system to unanticipated water hammer events. In the WCAPs, Westinghouse stated that there is a low potential for water hammer in the RCS and the connecting surge line, accumulator lines and RHR lines, since they are designed and operated to preclude voiding in the normally filled surge line and the auxiliary lines. Temperature during normal operation is maintained within a narrow range by the control rod positions; pressure is also controlled within a narrow range for steady state conditions by the pressurizer heaters and the pressurizer spray. Pressurizer safety and relief valve actuation and the associated hydraulic transients following valve opening are considered on the system design. Only relatively slow pressure transients are applicable to these lines and there is no significant effect on the system dynamic loads. In addition, As stated in the WCAPs, Westinghouse has instrumented reactor coolant systems to verify the flow and vibration characteristics of the system and the auxiliary lines. Licensees have performed pre-operational testing which, together with operating experience, has verified the design flow and vibration characteristics of the system and the auxiliary lines. Based on this, the NRC staff concludes that there is a low potential for the occurrence of unanticipated water hammer events in the surge line, the accumulator lines and the RHR lines because there is no evident source for such water hammers.

Low Cycle and High Cycle Fatigue

Draft SRP 3.6.3, Section III.9, also requires an assessment of the susceptibility of the piping system to low cycle and high cycle fatigue. Fatigue in piping systems resulting from normal operating thermal and mechanical transients are considered as low cycle fatigue. In the supplemental letter dated December 9, 2003, the licensee submitted the cumulative usage factors (CUFs) for the highest stressed locations in the three piping systems, and showed that these CUFs have met the limiting value of 1.0, in accordance with the rules for fatigue analysis of ASME Section III Class 1 piping.

High cycle fatigue in the piping systems is ordinarily caused by pump vibrations; Westinghouse stated that during operation, an alarm would typically signal the RCS pump vibration limits being exceeded. Field measurements on the surge line of typical PWR plants indicate stresses well below the fatigue endurance limit for the surge line material and would result in an applied stress intensity factor below the threshold for fatigue crack growth.

Based on this, the NRC staff concludes that in accordance with draft SRP 3.6.3, Section III.9, the licensee has demonstrated that the potential for pipe rupture due to thermal and mechanical induced fatigue is low. For the surge line, the licensee has also demonstrated that there is no potential for significant cyclic fatigue resulting from thermal stratification.

Internal Loads and Locations

The LBB fracture mechanics stability analysis or limit load evaluation requires the specification of the critical (highest stressed) locations for each system, and the applied internal loads at these locations. In accordance with draft SRP 3.6.3, Section III.10.c, the licensee listed, for each piping system and each location, the highest internal loads (forces, bending and torsional moments) under deadweight, operating pressure and thermal transients and the loads associated with safe shut down earthquake (SSE) conditions. The critical locations are determined based on the current licensing basis (CLB) design reports as having the least favorable combination of stress and material properties for base metal, weldments, and safe ends. The loads are combined in various load combinations for the assessment of leakage, critical crack size and crack stability analysis, as stipulated in Sections III.10.d, III.10.e, and III.10.f of draft SRP 3.6.3.

For the pressurizer surge line, the internal loads also included loads due to thermal stratification. The pressurizer surge line was previously evaluated for thermal stratification in conformance with the requirements of NRC Bulletin 88-11 [Reference 11], and the loads, accounting for thermal stratification, were described in WCAP-12893 and WCAP-12893, Supplement 1 [References 12 and 13]. These loads were used in the present LBB evaluation.

The staff finds the prescribed loads and critical locations for the accumulator and RHR lines acceptable since they were taken from the CLB design reports for these lines. The prescribed loads and critical locations for the surge line are also acceptable since they are based on the Westinghouse response to NRC Bulletin 88-11, which the staff has reviewed and accepted.

Conclusions

Based on the above evaluation, the NRC staff concludes that:

1. The normal operating loads and faulted condition loads for the three piping systems that form the basis for the LBB calculations are acceptable, since they were abstracted from design reports that form the current licensing basis for the Callaway plant

2. The justification for concluding that the probability for water hammer events is low in the three systems is acceptable, since it conforms with current industry experience.

3. The justification for concluding that the three systems will not be affected by low cycle fatigue, including fatigue due to thermal stratification in the surge line, or high cycle fatigue resulting from pump vibrations, is acceptable, since the licensee demonstrated that the fatigue evaluations meet the ASME Code Section III fatigue criterion.

4. The licensee has demonstrated that the snubber failure rates for the three systems will be maintained at a low level.

Based on these conclusions, the NRC staff further concludes that the three piping systems being considered for LBB in Section 3.2.1 of this SE meet the mechanical engineering criteria in draft SRP 3.6.3 and NUREG-1061, Volume 3.

3.2.2.2 Materials Engineering Evaluations

General Aspects of Licensee's LBB Analysis

The analyses provided by Westinghouse in the WCAP reports addressed the following four principal areas which are the criteria established for LBB analysis acceptability in NUREG-1061, Volume 3:

1. Demonstrate that the subject piping is a candidate for LBB analysis by showing that the piping is not particularly susceptible to active degradation mechanisms or atypical loading events.

2. Establish the critical through-wall flaw size under which analyzed locations would be expected to fail under normal operation (NOP) plus SSE or startup/shutdown loading conditions.

3. Establish the leakage behavior of smaller through-wall flaws under NOP loads alone for each location.

4. Evaluate the margin between the critical through-wall flaw size and an appropriate leakage through-wall flaw size and the stability of the through-wall leakage flaw.

Licensee's Evaluation of Pressurizer Surge, Accumulator, and RHR Lines

The LBB analysis of the pressurizer surge line, accumulator lines, and RHR piping that was submitted by the licensee as attachments to the licensee's June 27, 2003, letter was prepared by Westinghouse and given in WCAP-15983-P, WCAP-16019-P, and WCAP-16020-P, respectively. This section summarizes the results of the Westinghouse results for the four LBB criteria areas noted in Section 3.2.2.2 above.

In the discussion of the limitations of LBB analyses in NUREG-1061, Volume 3, it is stated that the LBB approach should not be considered when operating experience has indicated particular susceptibility to failure from the effects of corrosion, water hammer, or fatigue. Such mechanisms could cause the development of complex or extensive flaws in piping which significantly degrade its load carrying capacity while not propagating through-wall over a sufficient length to be detectable, or provide loads which are difficult to bound analytically. Westinghouse concluded that the pressurizer surge line, 10-inch accumulator lines, and 12-inch RHR piping have not been shown to be particularly susceptible to the effects of water hammer, intergranular stress corrosion cracking, erosion-corrosion, or creep.

As a result of the recent issue of primary water stress corrosion cracking (PWSCC) occurring in the V.C. Summer reactor vessel hot leg nozzle, Alloy 82/182 weld material is being currently investigated under the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) as susceptible to PWSCC. The licensee noted that the pressurizer nozzle safe-end-to-pipe weld location has an Alloy 82/182 weld and is included in the MRP program. Westinghouse stated in WCAP-15983-P that the results of the MRP showed that there was substantial margin between a leakage size flaw which would lead to a detectable leak and the size of flaw which would lead to pipe failure.

Regarding the potential for fatigue cracking from mechanical and thermal loadings, Westinghouse noted that low cycle fatigue considerations were accounted for in the design of this piping system through the fatigue usage factor evaluation to show compliance with the rules of Section III of the ASME Code (also in accordance with analyses in response to NRC Bulletin 88-11). Additionally, Westinghouse provided an analysis of the growth of postulated surface flaws based on design transient loading conditions and the analysis procedure suggested by Section XI, Appendix A of the ASME Code. Westinghouse showed that for semi-elliptic surface flaws with initial depths of up to (1) one-tenth of the pipe wall thickness for the pressurizer surge line (WCAP-15983) and (2) one-third of the pipe wall thickness for the accumulator and RHR lines (WCAPs 16019 and 16020), little or no growth was expected to occur. High cycle fatigue loads, primarily from pump vibrations, are managed through the monitoring of reactor coolant pump shaft vibration limits and inservice measurements have shown that the magnitude of the stresses associated with these vibrations is low and is not expected to raise a concern through the operating life of the facilities.

Next, the Westinghouse analysis (1) evaluated the piping by developing the applied stresses under NOP, NOP plus SSE, forced cooldown, and the low probability event of startup/shutdown plus SSE loading conditions, and (2) determined the leakage and critical through-wall flaw size for various locations along the piping. In the determination of the NOP applied stresses, the analysis included the tensile and bending stresses resulting from the internal pressure, deadweight, thermal expansion. For the pressurizer surge line, the licensee also provided another set of NOP loads which included the contribution of thermal stratification stresses at NOP conditions. SSE loads were added to the NOP loads (with and without NOP thermal stratification loads) when determining loads to be considered for the critical flaw size evaluation. In addition, forced cooldown and startup/shutdown plus SSE loading conditions were developed separately and are of interest for evaluating the allowable critical flaw size since large thermal bending stresses due to thermal stratification are developed in this piping as a result of the temperature difference between the hot leg and the pressurizer during these evolutions.

In the load combination, the deadweight, thermal expansion and/or thermal stratification, pressure, and SSE stresses were summed absolutely for the critical flaw size determination. Likewise, when evaluating the loads for the startup/shutdown conditions (with or without the SSE), the deadweight, thermal expansion, thermal stratification, and pressure loads were summed absolutely. The deadweight, thermal expansion and/or thermal stratification, and pressure stresses for NOP conditions were summed algebraically for the leakage flaw size determination.

For the purpose of LBB analysis, the critical flaw size can be defined as the longest preexisting through-wall flaw which could exist without growing unstably and result in a double-ended pipe rupture under faulted or off-normal loading conditions. This includes considering the NOP plus SSE and forced cooldown loading conditions. The analysis performed by Westinghouse to establish the critical flaw size at a nodal location was based on the use of a limit load analysis approach. This approach effectively predicts piping failure based on net section collapse of the cross-section which has been reduced by the through-wall cracked section. In the Westinghouse analysis, the SS welds were identified as the limiting material, i.e., the material for which the smallest margin between the critical and leakage flaw size exists. When analyzing SS welds using a limit load-based approach, an additional factor, the Z-factor, was incorporated to account for the generally lower toughness and lower load carrying capacity of SMAW welds. The Westinghouse analysis applied the Z-factor to increase the applied loads and thus reduce the through-wall flaw size which could be withstood without piping failure.

The leakage flaw size for an LBB analysis is defined as the flaw size which, under NOP conditions, would leak at a rate 10 times the amount of fluid detectable by the facility's containment leakage detection system. The factor of 10 is established in the LBB guidance of NUREG-1061, Volume 3 as the safety factor on leakage to account for uncertainties in calculating leakage from a through-wall crack. As noted in Section 5.2.3 of all three WCAPs, the performance of the Callaway leakage detection system is consistent with the guidelines of Regulatory Guide 1.45, and is capable of detecting a 1 gallon per minute (gpm) leak in one hour. Therefore, the leakage flaw calculated by Westinghouse at each nodal location was based on a leak rate of 10 gpm under NOP conditions. The leakage analysis performed by Westinghouse was based on the use of a Westinghouse proprietary methodology for calculating single or two-phase flow through cracks in light-water reactor piping.

In WCAP-15983-P for the pressurizer surge line, Westinghouse identified the limiting location to be node 3510 with a critical flaw size of 12.15 inches and leakage flaw size of 5.18 inches, and, therefore, this node has the minimum margin of 2.35 for a standard LBB evaluation. Likewise, in WCAP-16019-P for the accumulator lines, the limiting location is node 3295 that has the minimum margin of 2.64, and in WCAP-16020-P for RHR piping, the limiting nodal locations are Node 3285 (line loop 1) and Node 3020 (line loop 4), with margins of 4.37 and 3.01, respectively.

NRC Staff Evaluation

The NRC staff reviewed the scope of the licensee's LBB evaluation and concludes that the licensee adequately defined the analyzable segments of the piping system, as given above in Section 3.2.1 of this SE, for which LBB approval was sought.

The LBB analysis consists of a leakage flaw size calculation using loading associated with normal operating conditions and a critical flaw size calculation using loading associated with faulted conditions. The pipe loading associated with NOP conditions are axial forces and moments due to pressure, dead weight, and thermal expansion; and the pipe loading associated with faulted conditions are axial forces and moments of normal operating conditions in conjunction with SSE and seismic anchor motion loads. In the licensee's critical flaw size calculation, the absolute sum method was used to add the individual axial forces and moments into the combined axial forces and moments. Therefore, the recommended margin on loads of 1.0 is satisfied in accordance with draft SRP 3.6.3 (i.e., Criterion 2 of Section 3.2.2.2 of this SE).

Based on the material property, operating condition, and loading information noted in the foregoing discussion, the licensee implemented its LBB evaluation. This process first required determination of the leakage flaw size (i.e., the length of a through-wall circumferential flaw at the critical locations in the analyzed piping segments that would generate a leakage rate of 10 gpm (10 times the leakage detection capability of 1 gpm at Callaway). Therefore, a margin of 10 exists between the calculated leak rate from the leakage flaw and the leak detection capability of 1 gpm. The licensee then determined the critical flaw sizes for the critical locations that would be predicted to lead to piping failure under the faulted loading conditions. These critical flaw size calculations were performed using plots of limit moment versus crack lengths. The critical flaw size corresponds to the intersection of this curve and the maximum load line. The last step in the licensee's evaluation process was the calculation of ratios (margins) between the critical flaw size and the leakage flaw size for the critical locations. The relationship between the critical flaw size and leakage flaw size results from the criteria in draft SRP 3.6.3 and NUREG-1061, Volume 3, that specifies that a margin of two should be maintained for an acceptable LBB evaluation. In WCAP-15983-P, for the pressurizer surge line, Westinghouse identified the limiting location to be node 3510, with a critical flaw size of 11.42" and a leakage flaw size of 2.43" and, therefore, a margin of 2.43 for the standard LBB evaluation. Likewise, in WCAP-16019-P for the 10-inch accumulator lines, the limiting location is Node 3295 that has a margin of 2.64, and in WCAP-16020-P for 12-inch RHR piping, the limiting nodal locations are Node 3285 (line loop1) and Node 3020 (line loop 4), with margins of 4.37 and 3.01, respectively. Therefore, the margin on flaw size is acceptable. Based on the aforementioned discussion, all of the LBB recommended margins are satisfied. The NRC staff has determined that a postulated through-wall crack in the analyzed portion of pressurizer surge line, 10-inch accumulator lines and 12-inch RHR piping would remain stable and would not cause a gross failure of the component.

In regard to Criterion 1 in Section 3.2.2.2 of this SE, the LBB approach should not be considered when operating experience has indicated particular susceptibility to failure from the effects of corrosion, water hammer, or fatigue. The NRC staff agrees with the licensee's evaluation that water hammer should not occur in the subject piping because of system design, testing, and operational considerations. The licensee's evaluation of the effects of low and high cycle fatigue including the effects of thermal stratification were acceptable. The licensee addressed concerns regarding the impact of degradation mechanisms on the use of LBB for the subject lines. For the accumulator and RHR lines, stress corrosion cracking is precluded by use of fracture resistant materials (fine grain, solution annealed, controlled fabrication) in the piping system and controls on reactor coolant chemistry. Wall thinning by erosion-corrosion will not occur in the subject piping due to low velocity and use of austenitic stainless steel material.

For the pressurizer surge line, however, the pressurizer nozzle safe end to pipe weld has an Alloy 82/182 weld that is susceptible to PWSCC and, therefore, by Criterion 1, the NRC staff concludes that LBB is not applicable to this weld, and, therefore, not to the line. After discussions with the licensee on this weld, the licensee withdrew its request to apply LBB to this weld in its letter of April 5, 2004. In the letter, the licensee also requested that LBB be applied to the part of the pressurizer surge line that is downstream of the pipe whip restraints needed to hold the line in place for a double ended rupture of the line at this weld.

In its letter of April 5, 2004, the licensee stated that typically LBB is approved by the NRC staff for an entire line and that the analysis documented in WCAP-15983-P is for the entire pressurizer surge line. However, the licensee went on to state that its analysis assumed and evaluated a potential pipe break in the surge line at the Alloy 82/182 weld location and this analysis demonstrates that a break at this location does not cause a consequential failure at any other location along the pressurizer surge line. Additionally, the licensee stated that (1) it will continue to maintain all required plant equipment and structures, including pipe whip restraints, to address an assumed failure at this weld location, and (2) analyses currently being performed considering the plant configuration after the replacement steam generators are installed, will also consider a postulated break at this weld location.

Although the licensee has presented a technical basis that the pipe whip restraints on the pressurizer surge line that address the rupture of the line at the Alloy 82/182 weld location and the analyses that demonstrate no further rupture of the line if the line ruptures at the weld location, the NRC staff still has concerns about approving LBB on the portion of the line downstream of the pipe whip restraints needed to anchor that part of the line at the Alloy 82/182 weld location. The NRC staff has not approved an application of LBB for a segment of a piping system. To address the NRC staff's concerns, the licensee has been requested to demonstrate that effective mitigative measures are in place, or will be implemented, to counteract PSCC at the Alloy 82/182 weld. With such measures that are acceptable to the NRC staff, the NRC will then reconsider its concerns about applying LBB to this line.

Based on the above discussion, the NRC staff confirms the licensee's conclusion that the subject piping segments with the exception of the pressurizer surge line, can be shown to exhibit LBB behavior consistent with the criteria given in Section 3.2.2.2 of this SE, and in draft SRP 3.6.3 and NUREG-1061, Volume 3. This conclusion is based on the licensee's margins on leak rate, flaw size and combination of loads for stability of crack. The licensee's RCS leakage detection system is capable of detecting a leakage of 1 gpm in one hour. Based upon this information, the NRC staff concludes that LBB behavior has been demonstrated for the 10-inch accumulator lines and the 12-inch RHR piping.

Conclusions

Based on the materials engineering information and analyses provided by the licensee and discussed in Section 3.2.2.2 above, the NRC staff evaluated the LBB applicability of the analyzed portions of the 10-inch accumulator lines and the 12-inch RHR piping. Based on the above evaluation, the NRC staff concludes that because acceptable margins on leakage and crack size have been demonstrated, these sections of piping meet the LBB criteria in draft SRP 3.6.3 and, therefore, are consistent with the provisions of GDC 4, in that this piping will exhibit LBB behavior.

The pressurizer surge line is not approved for LBB applicability since the pressurizer surge line nozzle safe end to pipe weld contains Alloy 82/182 weld material and is susceptible to PWSCC and, therefore, by Criterion 1 in Section 3.2.2.2 of this SE, the NRC staff concludes that LBB is not applicable to this line at this time without effective mitigation measures. The licensee's revised request for application of LBB to the pressurizer surge line does not change the NRC staff's conclusion. If the licensee can demonstrate that effective mitigation measures acceptable to the NRC staff are in place or will be implemented to counteract PWSCC in the Alloy 82/182 weld, the NRC staff can approve this request at a later time.

3.2.2.3 LBB Conclusions

Based on the mechanical engineering and materials engineering evaluations of the LBB methodology being applied to the 10-inch accumulator lines and 12-inch RHR lines in Sections 3.2.2.1 and 3.2.2.2 of this SE, the NRC staff concludes the licensee is permitted to eliminate the dynamic effects associated with the postulated rupture of these sections of piping from the Callaway structural design basis.

For the pressurizer surge line, the NRC staff concluded in Section 3.2.2.1 of this SE that the line met the mechanical engineering criteria in draft SRP 3.6.3 and NUREG-1061, Volume 3; however, as stated in the conclusions of Section 3.2.2.2 of this SE, the line is not approved for LBB applicability at this time. In the letter approving this amendment, the licensee is being requested to demonstrate what effective mitigation measures are in place or will be implemented to counteract this mechanism. If these measures are acceptable to the NRC staff, then LBB can be approved for the line at a later time. The licensee was requested to provide the additional information within 60 days of the receipt of the letter.

3.3 Amendment Conclusion

The NRC staff reviewed the licensee's proposed amendment to determine the acceptability of the following changes to the licensing basis for the Callaway Plant:

1. A design modification to install a permanent access opening in the secondary shield wall to provide access to the sludge lance platforms around each SG in terms of occupational exposure.

2. Exclude the dynamic effects associated with a rupture of the 10-inch accumulator lines and the 12-inch RHR lines from the structural design basis for Callaway using LBB methodology pursuant to GDC 4.

Based on the conclusions of the NRC staff in Sections 3.1.4 and 3.2.2.3 of this SE on the design modification to the secondary shield wall and application of LBB to three piping systems, the NRC staff further concludes that the proposed amendment is acceptable except for the application of LBB for the pressurizer surge line and the ASCE 4-86 load combination methodology.

For the request to approve the ASCE 4-86 load combination methodology, the NRC staff has not completed its review of the methodology, but has identified the additional information it needs to complete its review in a conference call with the licensee on April 1, 2004.

For the pressurizer surge line, the pressurizer nozzle safe end to pipe weld has an Alloy 82/182 weld that is susceptible to PWSCC and LBB is not considered applicable to this line at this time. For the NRC staff to reconsider the acceptability of this line for LBB approval at a later time, the licensee is requested to demonstrate that effective mitigation measures that are acceptable to the NRC staff are in place or will be implemented to counteract this mechanism.

The amendment does not change the Callaway TSs. The amendment authorizes the licensee to include, in the FSAR, the application of LBB to exclude the dynamic effects of the 10-inch accumulator and the 12-inch RHR lines.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Missouri State official was notified of the proposed issuance of the amendment. The State official did not offer any comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents 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 (68 FR 43397). 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.

6.0 CONCLUSION

The Commission 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from D. Shafer, AmerenUE, to the NRC Document Control Desk, with attached Westinghouse Electric Company, LLC, Proprietary Reports, "Application of Proprietary Leak-Before-Break (LBB) Methodology Reports and Draft Regulatory Guide DG-1108," June 27, 2003.

2. Letter from K. D. Young, AmerenUE, to the NRC Document Control Desk, "Application of Proprietary Leak-Before-Break (LBB) Methodology Reports and Draft Regulatory Guide DG-1108," Responses to Requests for Additional Information, December 9, 2003.

3. Letter (ULNRC-04939) from Keith D. Young, AmerenUE, to the NRC Document Control Desk, "Application of Proprietary Leak-Before-Break (LBB) Methodology Reports and Draft Regulatory Guide DG -1108," January 14, 2004.

4. Letter (ULNRC-04974) from Keith D. Young, AmerenUE, to the NRC Document Control Desk, "Application of Proprietary Leak-Before-Break (LBB) Methodology Reports and Draft Regulatory Guide DG -1108," April 5, 2004.

5. Westinghouse Electric Company, LLC, Proprietary Report WCAP-15983-P, Revision 0, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," February 2003.

6. Westinghouse Electric Company, LLC, Proprietary Report WCAP-16019-P, Revision 0, "Technical Justification for Eliminating 10" Accumulator Lines Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," February 2003.

7. Westinghouse Electric Company, LLC, Proprietary Report WCAP-16020-P, Revision 0, "Technical Justification for Eliminating 12" Residual Heat Removal (RHR) Lines Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," February 2003.

8. NUREG-0800, Standard Review Plan, Draft Section 3.6.3, "Leak-Before Break Evaluation Procedures," 52 FR 32626-32633, August 28, 1987.

9. NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," November 1984.

10. USNRC Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," dated February 1, 1984.

11. USNRC Bulletin No. 88-11, "Pressurizer Surge Line Thermal Stratification," December 20, 1988.

12. Westinghouse Electric Company, LLC, Proprietary Report WCAP-12893, "Structural Evaluation of the Wolf Creek and Callaway Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," March 1991.

13. Westinghouse Electric Company, LLC, Proprietary Report WCAP-12893, Supplement 1, "Structural Evaluation of the Wolf Creek and Callaway Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," December 1995.

14. USNRC Draft Regulatory Guide DG-1108, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," proposed Revision 2 of Regulatory Guide 1.92, August 2001.

15. American Society of Civil Engineers, Standard 4-86, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety Related Nuclear Structures," September 1986.

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Date: April 12, 2004

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

UNION ELECTRIC COMPANY

DOCKET NO. 50-483

NOTICE OF PARTIAL WITHDRAWAL OF APPLICATION FOR

AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has granted the request of Union Electric Company (the licensee) to partially withdraw its June 27, 2003, application for proposed amendment to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1, located in Callaway County, Missouri.

The proposed amendment will approve the application of leak-before-break methodology for the accumulator and residual heat removal lines and installation of an opening the secondary shield wall in terms of the effect of the opening on occupational exposure. The shield wall opening is related to plant modifications that would facilitate maintenance on the replacement steam generators to be installed in Refueling Outage (RO) 14 (Fall 2005). The licensee withdrew the part of the amendment request that would apply LBB to the pressurizer surge line Alloy 82/182 weld location.

The Commission had previously issued a Notice of Consideration of Issuance of Amendment published in the *Federal Register* on July 22, 2003 (68 FR 43397). However, by letter dated April 5, 2004, the licensee partially withdrew the proposed change.

For further details with respect to this action, see the application for amendment dated June 27, 2003, and the licensee's letter dated April 5, 2004, which partially withdrew the application for license amendment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams/html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1-800-397-4209, or 301-415-4737 or by email to pdr@nrc.gov.

Dated at Rockville, Maryland, this 12th day of April 2004.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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