Union Electric

One Ameren Plaza
1901 Chouteau Avenue
PO Box 66149
St. Louis, MO 63166-6149
314.621.3222

June 27, 2003

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

Ladies and Gentlemen: ULNRC-04868

DOCKET NUMBER 50-483
UNION ELECTRIC COMPANY
CALLAWAY PLANT
APPLICATION OF PROPRIETARY LEAK-BEFORE-BREAK (LBB)
METHODOLOGY REPORTS AND DRAFT REGULATORY GUIDE DG-1108

AmerenUE herewith transmits an application for amendment to Facility Operating License No. NPF-30 for the Callaway Plant.

In order to facilitate maintenance on the replacement steam generators (SGs) to be installed during Refuel 14 (fall of 2005), the existing sludge lance platforms will be replaced during Refuel 13 (spring 2004) with new platforms that will provide a larger area around each SG. As part of this Refuel 13 modification, a permanent access opening will be cut through the secondary shield wall at the 'C' loop SG to allow direct access to the sludge lance platforms from the 2026' elevation level. To support the secondary shield wall access opening modification, NRC review and approval of the following are requested:

1. Dynamic effects associated with large RCS branch line ruptures (pressurizer surge line, accumulator lines, and RHR lines) may be excluded from the design basis using the Leak-Before-Break (LBB) methodology, provided NRC approval of this methodology's application is obtained pursuant to General Design Criterion (GDC) 4 of 10 CFR 50, Appendix A. LBB topical reports are attached for NRC review and approval, as discussed below.

2. The use of the ASCE 4-86 "100-40-40" method for combining the three orthogonal components of seismic response loads, discussed in Draft Regulatory Guide DG-1108 (proposed Revision 2 of RG 1.92), can be used in lieu of the current square-root-sum-of-the-squares (SRSS) method, provided NRC approval of this methodology change is obtained pursuant to 10 CFR 50.59(c)(2)(viii) and 10 CFR 50.90. A001

Attached to this letter are copies of the following proprietary topical reports for NRC review and approval:

> WCAP-15983-P, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant" (Proprietary);
>
> WCAP-16019-P, "Technical Justification for Eliminating 10" Accumulator Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant" (Proprietary); and
>
> WCAP-16020-P, "Technical Justification for Eliminating 14" Residual Heat Removal (RHR) Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant" (Proprietary) dated February 2003.

This letter also transmits non-proprietary copies of the above topical reports. Westinghouse has determined that information associated with WCAP-15983-P, WCAP-16019-P, and WCAP-16020-P is proprietary, and is thereby supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.790. Accordingly, it is respectfully requested that the information that is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.790.

Also enclosed are Westinghouse authorization letter CAW-03-1606, its accompanying affidavit, Proprietary Information Notice, and Copyright Notice. Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-03-1606 and should be addressed to H.A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Attachments 1, 2, and 3 provide the Evaluation, Topical Reports, and Proposed FSAR changes, respectively, in support of this application. Attachment 3 mark-ups are provided for information only, with the intention of facilitating NRC review of this amendment application. Some of the FSAR mark-ups can not be finalized until the final replacement SG loads have been determined. Final FSAR changes will be implemented after this amendment is approved, subject to the updating requirements of 10CFR50.71(e). No other new commitments are contained in this amendment application.

It has been determined that this amendment application does not involve a significant hazards consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental assessment need be prepared in connection with the issuance of this amendment.

NRC approval of this amendment application is requested by February 1, 2004 so that the amendment and its associated plant modifications can be implemented prior to entering MODE 4 ascending during startup from Refuel 13, currently scheduled for spring 2004. In accordance with 10CFR50.91, a copy of this amendment application is being provided to the designated Missouri State official. If you have any questions on this amendment application, please contact us.

Very truly yours,

David Shafer
Acting Manager, Regulatory Affairs

DS/GGY/mlo

Attachments:
1) Evaluation
2) a. Topical Reports (Proprietary)
   b. Topical Reports (Non-proprietary)
   c. Proprietary Affidavit
3) Proposed FSAR Changes (for information only)

STATE OF MISSOURI)
) S S
CITY OF ST. LOUIS)

David Shafer, of lawful age, being first duly sworn upon oath says that he is Acting Manager, Regulatory Affairs, for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By David Shafer
David Shafer
Acting Manager, Regulatory Affairs

SUBSCRIBED and sworn to before me this 27th day of June, 2003.

Carol A. Head

CAROL A. HEAD
Notary Public - Notary Seal
STATE OF MISSOURI
St. Charles County
My Commission Expires: Sept. 23, 2006

cc: U. S. Nuclear Regulatory Commission (Original) w/a
Attn: Document Control Desk
Mail Stop P1-137
Washington, DC 20555-0001

Mr. Thomas P. Gwynn w/o
Regional Administrator
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

Senior Resident Inspector w/o
Callaway Resident Office
U.S. Nuclear Regulatory Commission
8201 NRC Road
Steedman, MO 65077

Mr. Jack N. Donohew (2 copies) w/a
Licensing Project Manager, Callaway Plant
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop 7E1
Washington, DC 20555-2738

Manager, Electric Department w/o
Missouri Public Service Commission
PO Box 360
Jefferson City, MO 65102

Mr. Gerhard K. Samide w/o
ANI Account Engineer
Town Center, Suite 3005
29 S. Main St.
West Hartford, CT 06107-2445

Deputy Director w/o
Department of Natural Resources
Central Regional Office
P.O. Box 176
Jefferson City, MO 65102-0176

Ms. Jennifer L. Yunk
Superintendent, Licensing
Wolf Creek Nuclear Operating Corporation
PO Box 411
Burlington, KS 66839

EVALUATION

EVALUATION

1.0 DESCRIPTION

The proposed amendment would allow the use of different methodologies for determining the required loads on the 'C' loop steam generator (SG) cubicle secondary shield wall at Callaway.

2.0 PROPOSED CHANGE

The proposed amendment is needed to facilitate maintenance on the replacement SGs to be installed during Refuel 14 (fall of 2005). The existing sludge lance platforms will be replaced during Refuel 13 (spring 2004) with new platforms that will provide a larger area around each SG. As part of this Refuel 13 modification, a permanent access opening will be cut through the secondary shield wall at the 'C' loop SG to allow direct access to the sludge lance platforms from the 2026' elevation level. To support the secondary shield wall access opening modification, NRC review and approval of the following are requested:

1. Dynamic effects associated with large RCS branch line ruptures (pressurizer surge line, accumulator lines, and RHR lines) may be excluded from the design basis using the Leak-Before-Break (LBB) methodology, provided NRC approval of this methodology's application is obtained pursuant to General Design Criterion (GDC) 4 of 10 CFR 50, Appendix A. LBB topical reports are submitted in Attachment 2 of this amendment application for NRC review and approval.

2. The use of the ASCE 4-86 "100-40-40" method for combining the three orthogonal components of seismic response loads, discussed in Draft Regulatory Guide DG-1108 (proposed Revision 2 of RG 1.92), can be used in lieu of the current square-root-sum-of-the-squares (SRSS) method, provided NRC approval of this methodology change is obtained pursuant to 10 CFR 50.59(c)(2)(viii) and 10 CFR 50.90.

Attachments 2 and 3 provide the topical reports and proposed FSAR changes (latter is for information-only) associated with this amendment application.

3.0 BACKGROUND

The reinforced concrete secondary shield walls are 3 feet - 6 inches thick and are anchored to the reactor building base slab. The walls extend from the base slab to a level above the top of the SG tube bundle to provide shielding for the reactor coolant system. Portions of the secondary shield walls above the operating floor are designed to be removable for SG removal. These reinforced concrete wall panels are bolted together at vertical joints to provide for structural continuity and integrity. They are keyed into the slab at the bottom of the panels and are prevented from becoming missiles during a seismic event. The secondary shield walls, in conjunction with the primary shield wall and refueling canal walls, form the loop compartments and provide support for the SGs, reactor coolant pumps, pressurizer, cross-over legs, piping, various equipment, platforms, and elevated floors.

The secondary shield walls are currently designed to resist all of the applicable loads, including those due to differential pressure and temperature resulting from a LOCA, RCS component support forces, operating basis earthquake (OBE) and safe shutdown earthquake (SSE), dead and live loads from the operating floor and intermediate platforms and walkways, and those loads resulting from a postulated pipe break. Analysis of the secondary shield walls is performed using classical techniques and the SAP computer program described in FSAR Appendix 3.8A. Design for the effects of postulated pipe breaks is performed using Bechtel topical report BN-TOP-2. The finite element model used for analyzing the secondary shield walls consists of a three-dimensional model of one-half of the structure in plan about an axis of symmetry. An additional finite element model is used for analyzing these secondary shield walls at the pressurizer. Appropriate boundary conditions are modeled to simulate actual conditions at the axis of symmetry and at the intersections with the base slab, refueling canal walls, floors, and RCS component supports. The analysis for time-dependent loads, such as those for differential pressure and pipe rupture reaction forces, is currently performed in a manner similar to that used for the primary shield wall discussed in FSAR Section 3.8.3.4.2. Design of the secondary shield walls is performed using the strength design methods described in ACI 318-1971.

During Refuel 13 (scheduled for the spring of 2004), the following design changes are planned:

1. Replacement of the existing sludge lance platforms, which are small and uneven, with new platforms that will provide a larger area around each SG and at a uniform level to facilitate movement of equipment and tool boxes.

2. Installation of a walkway between the 'A' loop and 'B' loop reactor coolant pumps (RCPs) that will tie the 'A/D' and 'B/C' SG cubicle platforms together.

3. Installation of a new ladder that will provide access from the 2000' elevation to the new walkway in the vicinity of the 'B' loop RCP.

4. Rotation of the existing primary to sludge lance platform ladders 90°.

5. Addition of a permanent opening (approximately 26 inches wide by 82 inches tall) in the secondary shield wall at the 'C' loop SG and an access platform to allow direct access to the sludge lance platforms from the 2026' elevation level. Selection of this spot for the access opening was done to minimize the relocation of existing plant equipment, small bore lines, supports, tubing, conduit, junction boxes, etc.

6. Relocation of piping and electrical equipment that are interferences to installation of the new platforms and secondary shield wall opening.

The opening in the 'C' loop SG secondary shield wall will allow direct access to this SG cubicle, and via use of the continuous SG sludge lance platforms that extend to the other SG cubicles, will provide inside secondary wall access to those areas as well. This is a significant improvement over the current method of accessing the SG cubicle areas from outside of the secondary shield wall, which involves the difficulties of moving personnel and equipment over various elevation differences to gain access to the SG cubicle areas to perform SG inspection and maintenance activities.

From inside and outside the 'C' loop SG secondary shield wall (or cubicle wall), the access opening will be made by removing a rectangular section of concrete in the southeast cubicle wall (oriented to plant north) near that wall's juncture with the east wall of the cubicle. Use of precision cutting equipment will allow adapting the geometries of the opening to optimize maintaining the strength and integrity of the cubicle walls, as well as minimizing the need to relocate piping/tubing and electrical conduit currently located and supported off of the inner and outer cubicle wall surfaces in the area of the concrete cut.

Cutting of the concrete section through the 'C' loop SG secondary shield wall will be accomplished first by drilling bore holes and then using diamond wire concrete cutting machines to make several cuts to enable removal of the concrete section in more easily handled pieces. The secondary shield wall drilling and cutting activities will both be performed by a qualified vendor for such services, and employ processes designed to control and contain the resultant debris for appropriate disposal from containment. Heavy load handling of

equipment and concrete sections resulting from cutting activities will be controlled per plant procedure MDP-ZZ-MH004 in accordance with plant commitments made in responses to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and related NRC Generic Letters and Bulletin 96-02.

With the completion of concrete cutting, the exposed opening surfaces will be cleaned and coated. The opening will be provided with a caged entryway with a lockable gate for access control. In order to preclude radiation streaming and dose resulting from creating the opening in the secondary shield cubicle wall, alternative shielding will be applied to the opening and access control entryway to limit radiation doses consistent with maintaining them as low as reasonably achievable (ALARA).

Item 5 above (cutting permanent access opening) requires prior NRC review and approval of the two methodology changes discussed in this amendment application before it can be implemented.

NRC review and approval of the LBB topical reports will allow the exclusion of the dynamic effects of specific primary side branch line ruptures, and as a result will reformulate the SG lateral support loads that are transferred to the 'C' loop SG cubicle secondary shield wall where the permanent access opening will be located. Transferred loadings to the 'C' loop SG cubicle secondary shield wall resulting from small piping breaks and seismic loadings will remain in the design basis of this secondary shield wall. Engineering evaluations conclude that these design basis functions will still be met with the permanent access opening present within the 'C' loop SG cubicle secondary shield wall. Those evaluations considered the loadings of the secondary shield wall itself, dynamic loadings to the shield wall from supported structures, systems, and components (SSCs) for small pipe breaks and their associated pressure effects within the 'C' loop SG cubicle, and seismic interaction loadings.

NRC review and approval of the use of the "100-40-40" method for combining orthogonal components of seismic excitation loads for the 'C' loop SG cubicle secondary shield wall access opening is requested as an alternate to the square-root-sum-of-the-squares (SRSS) method such that the sign convention of the applied loads can be maintained. Resulting stresses using the "100-40-40" method remain within the allowable code limits of ACI 318-1971.

4.0 TECHNICAL ANALYSIS

Leak-Before-Break Topical Reports

NRC regulations require that licensees provide protective measures against the dynamic effects of postulated pipe breaks in high energy fluid system piping, except where NRC approval per GDC 4 has been obtained as discussed in Section 5.2 herein. Protective measures include physical isolation from postulated pipe rupture locations, if feasible, or the installation of pipe whip restraints, jet impingement shields, or compartments.

In 1975, concerns arose as to the asymmetric loads on pressurized water reactor (PWR) vessels and their internals which could result from large postulated breaks at discrete locations in the main primary coolant loop piping. This led to the establishment of Unresolved Safety Issue (USI) A-2, "Asymmetric Blowdown Loads on PWR Primary Systems," as discussed in NUREG-0609. Advanced fracture mechanics technology was applied in topical reports (WCAP-9558, WCAP-9787, Letter Report NS-EPR-2519) submitted to the staff by Westinghouse on behalf of the licensees belonging to the USI A-2 Owners Group. Although the topical reports were intended to resolve the issue of asymmetric blowdown loads that resulted from a limited number of discrete break locations, the technology advanced in these topical reports demonstrated that the probability of breaks occurring in the primary coolant system main loop piping is sufficiently low that these breaks need not be considered as a design basis for requiring installation of pipe whip restraints or jet impingement shields. The NRC staff, after several review meetings with the Advisory Committee on Reactor Safeguards (ACRS) and a meeting with the NRC Committee to Review Generic Requirements (CRGR), concluded that an exemption from the version of GDC 4 at that time would be acceptable as an alternative for resolution of USI A-2. This NRC staff position was stated in Generic Letter 84-04 (Reference 7.1). Later that year, NUREG-1061 Volume 3 (Reference 7.2) was issued with guidance on the application, including margin requirements, of the underlying concept which came to be referred to as leak-before-break (LBB).

The acceptance of an exemption to GDC 4 at that time, and ultimately a revision to GDC 4, was made possible by the development of advanced fracture mechanics technology. These advanced fracture mechanics techniques deal with relatively small flaws in piping components (either postulated or real) and examine their behavior under various pipe loads. The objective is to demonstrate by deterministic analyses that the detection of small flaws by inservice inspection or leakage monitoring systems is assured long before the flaws can grow to critical or unstable sizes which could lead to large break areas such as the double-ended guillotine break (DEGB) or its equivalent. There is no implication

that piping failures cannot occur, but rather that improved knowledge of the failure modes of piping systems and the application of appropriate remedial measures, if indicated, can reduce to insignificant values the probability of catastrophic failure.

Since 1984, various methodology refinements have occurred. Guidance in Standard Review Plan (SRP) Draft Section 3.6.3 (Reference 7.3) was used to develop the topical reports enclosed herein as Attachment 2 (WCAP-15983, WCAP-16019, and WCAP-16020).

The following conclusions can be drawn after reviewing WCAP-15983, WCAP-16019, and WCAP-16020:

1. The Westinghouse reactor coolant system (RCS) Class 1 lines have an operating history that demonstrates their inherent operating stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking, IGSCC), water hammer, or fatigue (low and high cycle). This operating history totals more than 1100 reactor-years.

2. Stress corrosion cracking is precluded by use of fracture resistant materials in the piping system and controls on reactor coolant chemistry, temperature, pressure, and flow during normal operation.

   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 is being currently investigated under the EPRI Materials Reliability Project (MRP) Program. It should be noted that the pressurizer nozzle safe end to pipe weld location at Callaway has an Alloy 82/182 weld and is included in the EPRI MRP Program. The results of the MRP Program show that there is a substantial margin between the size flaw which would lead to a detectable leak and the size flaw which could lead to failure. The susceptible material under EPRI investigation is not found in the RHR or accumulator lines evaluated in the above topical reports for Callaway.

3. There is a low potential for water hammer in the RCS primary loop piping and attached Class 1 branch lines because of system design, testing, and operational considerations that preclude voiding in the normally filled RCS and branch line piping.

4. The effects of low and high cycle fatigue on the integrity of the pressurizer surge line, accumulator lines, and RHR lines were evaluated and shown to be acceptable. Thermal stratification, thermal aging, and creep damage

were shown to not be applicable to the accumulator lines and RHR lines. The pressurizer surge line is subjected to thermal stratification and the effects of stratification can be significant during certain modes of heatup and cooldown operation. The effects of stratification have been evaluated for Callaway's surge line and the loads, accounting for the stratification effects, have been derived in WCAP-12893 (Reference 7.4) and WCAP-12893 Supplement 1 (Reference 7.5). These loads are used in the LBB evaluation described in Section 4.4 of WCAP-15983 (see Attachment 2). The Callaway surge line piping system is fabricated from forged products (see Section 3 of WCAP-15983) which are not susceptible to toughness degradation due to thermal aging. Finally, the maximum operating temperature of the pressurizer surge line piping, which is about 650°F, is well below the temperature that would cause any creep damage in stainless steel piping. Cleavage type failures are not a concern for the operating temperatures and the material used in the stainless steel piping of the pressurizer surge line.

5. Ample margin exists between the leak rate of small stable flaws and the capability of the Callaway RCS pressure boundary leakage detection system. Callaway has an RCS pressure boundary leak detection system which is consistent with the guidelines of Regulatory Guide (RG) 1.45, and the system can detect leakage of 1 gpm in 1 hour. The leak rate through the postulated leakage-size flaw results in a factor of at least 10 relative to the sensitivity of the Callaway leak detection system.

6. Ample margin exists between the small stable leakage flaw sizes of item 5 above and the critical flaw size. The minimum margin between the 10-gpm leakage-size flaw and the critical-size flaw exceeds the required margin of 2, as follows:

   3.01 (RHR lines):

   2.64 (accumulator lines); and

   2.35 (pressurizer surge line).

   Tables 7-1 and 7-2 in each of the Attachment 2 topical reports discuss the application of margins and how each requirement is met.

7. At the critical weld locations, the 10-gpm leakage-size flaws were shown to be stable using the faulted loads obtained by the <u>absolute sum</u> method per draft SRP 3.6.3 section III.10.f. Therefore, the margin on loads of 1.0 using the <u>absolute summation</u> of faulted load combinations is satisfied.

8. Representative minimum yield strength and minimum ultimate strength at operating temperature were used for the flaw stability evaluations. Representative average yield strength properties were used for the leak rate calculations. See Section 3 and Table 3-2 in each of the Attachment 2 topical reports for a discussion of material properties. Conservative "Z" correction factors from Reference 7.3 were applied in the stability evaluations.

9. Fatigue crack growth was demonstrated to not be a concern for the RHR lines, accumulator lines, and pressurizer surge line.

In conclusion, the postulated reference flaw will be stable because of the ample margins and will leak at a detectable rate which will assure a safe plant shutdown. Based on the above, it is concluded that pressurizer surge line breaks, 10-inch accumulator line breaks, and 12-inch RHR line breaks should not be considered in the structural design basis of Callaway Plant.

<u>Seismic Load Combination Methodology</u>

Callaway is currently committed to Revision 1 of RG 1.92 to the extent discussed in FSAR Appendix 3A. For the design of the secondary shield walls, the discussion in FSAR Section 3.7(B).2.6 cites two Bechtel Power Corporation topical reports that are incorporated by reference per FSAR Table 1.6-1. The Bechtel topical reports discuss the use of SRSS for combining the three orthogonal components of seismic response loads. This is consistent with Regulatory Position C.2.1 of RG 1.92, Revision 1.

However, an alternate method of combining the three spatial components of earthquake loads is discussed in Draft RG DG-1108, which is the staff's proposed Revision 2 to RG 1.92 (Reference 7.6). DG-1108 has been changed from Revision 1 to endorse the "100-40-40" percent combination rule of the American Society of Civil Engineers in ASCE Standard 4-86 (Reference 7.7) which preserves the mathematical sign when it is necessary to distinguish direction. The "100-40-40" percent rule was originally proposed as a simple way to estimate the maximum expected response of a structure subject to three-directional seismic loading, for response spectrum analysis.

For response spectrum analysis when each of the three spatial components are calculated independently, the SRSS procedure for combining the values of the response to the three components of an earthquake is based on the consideration that it is very unlikely that peak values of a response of a given element would occur at the same time during an earthquake. That is, the acceptance of the method of SRSS is based on the assumption of uncorrelated

seismic ground motions. The results of SRSS spatial combination have been compared with the "100-40-40" spatial combination. Generally the "100-40-40" combination method produces higher estimates of maximum response than the SRSS combination method; however, the "100-40-40" method more realistically reflects the actual physical loading on the secondary shield walls when it is necessary to maintain the directional-indicating mathematical sign that would be lost in taking a square root of the sum of the squares of the loads in the SRSS method.

In accordance with ASCE Standard 4-86, "Seismic Analysis of Safety-Related Nuclear Structures" (Reference 7.7), when the response spectra method is adopted for seismic analysis of uncorrelated seismic ground motions, the representative maximum values of the structural responses to each of the three components of earthquake motion may be combined using the "100-40-40" method of combination in lieu of the SRSS method. The "100-40-40" method is as follows:

1. Let R_1, R_2, R_3 be the maximum co-directional responses caused by each of the three components of earthquake at a particular point of the structure, such that

$$|R_1| \geq |R_2| \geq |R_3|$$

2. The maximum seismic response, R_{max}, that is due to simultaneous earthquake loading in three directions is given by

$$R_{max} = \pm (1.0 * R_1 + 0.4 * R_2 + 0.4 * R_3)$$

   Note that equation 17 in Regulatory Position C.2.1 of DG-1108 contains typographical errors. No absolute value signs should be used in equation 17 since the objective of the "100-40-40" method is to preserve the mathematical sign convention of the applied loads.

As its name implies the SRSS method would yield:

$$R = (R_1^2 + R_2^2 + R_3^2)^{1/2}$$

The structural analysis performed to date for the 'C' loop SG secondary shield wall after cutting the permanent access opening, when using the SRSS method

and assuming that dynamic effects can be excluded from the successful application of the LBB methodology discussed above, indicate that all concrete and reinforcing steel stresses, including reinforcing steel development length, still meet the applicable ACI 318-71 acceptance criteria for the existing SG loads. The 'C' loop SG secondary shield wall has been analyzed for the appropriate load combinations described in FSAR Section 3.8.3.3. The results of this structural analysis were evaluated against the acceptance criteria in ACI 318-71 and remain acceptable. NRC approval to use the ASCE methodology will allow for more realistic seismic loading combinations when the structural analysis is revised to account for the final replacement SG loads which are still being determined, but are expected to be slightly higher.

<u>Summary/Conclusion</u>

The discussions presented above assess the potential impact of this amendment and demonstrate that the proposed amendment will not adversely affect the design basis or the safe operation of the plant.

5.0 REGULATORY SAFETY ANALYSIS

This section addresses the standards of 10CFR50.92 as well as the applicable regulatory requirements and acceptance criteria.

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

The proposed amendment would allow the use of different methodologies for determining the required loads on the secondary shield walls at Callaway. NRC review and approval of the following are requested:

1. Dynamic effects associated with large RCS branch line ruptures (pressurizer surge line, accumulator lines, and RHR lines) may be excluded from the design basis using the Leak-Before-Break (LBB) methodology, provided NRC approval of this methodology's application is obtained pursuant to General Design Criterion (GDC) 4 of 10 CFR 50, Appendix A. LBB topical reports are submitted in Attachment 2 of this amendment application for NRC review and approval.

2. The use of the ASCE 4-86 "100-40-40" method for combining the three orthogonal components of seismic response loads, discussed in Draft Regulatory Guide DG-1108 (proposed Revision 2 of RG 1.92), can be used in lieu of the current square-root-sum-of-the-squares (SRSS) method, provided NRC approval of this methodology change is obtained pursuant to 10 CFR 50.59(c)(2)(viii) and 10 CFR 50.90.

The proposed amendment does not involve a significant hazards consideration for Callaway Plant based on the three standards set forth in 10CFR50.92(c) as discussed below:

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

Response: No

Overall protection system performance will remain within the bounds of the previously performed accident analyses. The design of the protection systems will be unaffected. The reactor protection system and engineered safety feature actuation system will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

Neither the currently intact 'C' SG cubicle secondary shield wall, nor the proposed configuration that provides a permanent access opening, create accident initiation mechanisms that would increase the probability of an accident. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed amendment will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR.

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

(2) Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

There are no changes in the method by which any safety-related plant system performs its safety function. This amendment will not affect the normal method of power operation or change any operating parameters. No performance requirements will be affected, but SG maintenance access will be greatly improved.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment.

Presence of a permanent access opening in the 'C' loop SG secondary shield wall does not, of itself, create the possibility of a new accident since the secondary shield walls are not used for missile protection and the high-energy line breaks (greater than 10-inches in diameter) that would generate missiles will be removed from the structural design basis after NRC's review and acceptance of the LBB topical reports.

The proposed amendment does not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems.

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

(3) Does the proposed change involve a significant reduction in a margin of safety?

Response: No

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss of coolant accident peak cladding temperature (LOCA PCT), or peak local power density. The LBB margins discussed in NUREG-1061 Volume 3 are satisfied. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met. The secondary shield walls are not fission product barriers. They provide radiation shielding to maintain occupational exposure ALARA and provide structural support to primary coolant system SSCs.

The proposed amendment does not eliminate any surveillances or alter the Frequency of surveillances required by the Technical Specifications. The nominal Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) trip setpoints (TS Bases Tables B 3.3.1-1 and B 3.3.2-1), RTS and ESFAS allowable values (TS Tables 3.3.1-1 and 3.3.2-1), and the safety analysis limits assumed in the transient and accident analyses (FSAR Table 15.0-4) are unchanged. None of the acceptance criteria for any accident analysis is changed.

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

<u>Conclusion</u>:

Based on the above, AmerenUE concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The regulatory bases and guidance documents associated with the systems discussed in this amendment application include:

GDC 2 requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without the loss of the capability to perform their safety functions.

GDC 4 requires that structures, systems, and components important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with the normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, discharging fluids that may result from equipment failures, and from events and conditions outside the nuclear power unit. However, 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.

GDC 16 requires that the reactor containment and associated systems be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

GDC 50 requires that the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in SGs and as required by 10 CFR 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

GDC 51 through GDC 57 cover various design and testing requirements for the reactor containment structure, including fracture prevention, leak rate testing, and isolation provisions that are dependent on the type of containment penetration.

NUREG-0800, Standard Review Plan, Draft Section 3.6.3, "Leak-Before-Break Evaluation Procedures," 52 FR 32626-32633, August 28, 1987, was used to develop WCAP-15983, WCAP-16019, and WCAP-16020. This guidance has been cited in previous NRC staff safety evaluations of LBB applications.

Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Proposed Revision 2, August 2001 (Draft Regulatory Guide DG-1108), endorses the "100-40-40" combination rule of the American Society of Civil Engineers (ASCE) Standard 4-86 (Reference 7.7). This rule preserves the mathematical sign when it is necessary to distinguish direction.

Various other NRC Regulatory Guides (i.e., RG 1.10, RG 1.15, RG 1.55, RG 1.69, and RG 1.94) are applicable to the design and construction of the reactor building internal structures. Specific editions and the extent of compliance with these guides is discussed in FSAR Appendix 3A.

There will be no changes such that compliance with any of the regulatory requirements and guidance documents above would come into question. The evaluations performed by AmerenUE confirm that Callaway Plant will continue to

comply with all applicable regulatory requirements. GDC 16 and GDC 50 through 57 are mentioned here for reference only since containment subcompartments (SG loop and pressurizer vault) are involved. The secondary shield walls perform none of the radioactivity containment functions covered by these design criteria. The secondary shield walls provide radiation shielding to maintain occupational exposure ALARA and provide structural support to primary coolant system SSCs.

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

6.0 ENVIRONMENTAL CONSIDERATION

AmerenUE has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10CFR20, or would change an inspection or surveillance requirement. However, AmerenUE has evaluated the proposed amendment and has determined that the amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22 (c)(9). Therefore, pursuant to 10CFR51.22 (b), an environmental assessment of the proposed amendment is not required.

7.0 REFERENCES

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

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

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

7.4. WCAP-12893, "Structural Evaluation of the Wolf Creek and Callaway Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," March 1991 (Westinghouse Proprietary).

7.5. WCAP-12893, Supplement 1 "Structural Evaluation of the Wolf Creek and Callaway Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," December 1995 (Westinghouse Proprietary).

7.6. Draft Regulatory Guide DG-1108 (Proposed Revision 2 of Regulatory Guide 1.92), "Combining Modal Responses and Spatial Components in Seismic Response Analysis," August 2001.

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

ATTACHMENT TWO

WESTINGHOUSE TOPICAL REPORTS

a. Topical Reports (Proprietary)

1. Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant

2. Technical Justification for Eliminating 10" Accumulator Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant

3. Technical Justification for Eliminating 12" Residual Heat Removal (RHR) Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant

b. Topical Reports (Non-proprietary)

1. Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant

2. Technical Justification for Eliminating 10" Accumulator Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant

3. Technical Justification for Eliminating 12" Residual Heat Removal (RHR) Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant

c. Proprietary Affidavit

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

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

Direct tel: (412) 374-5282
Direct fax: (412) 374-4011
e-mail: Sepplha@westinghouse.com

Our ref: CAW-03-1606

March 13, 2003

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-15983-P, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant" (Proprietary)
WCAP-16019-P, "Technical Justification for Eliminating 10" Accumulator Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant" (Proprietary)
WCAP-16020-P, "Technical Justification for Eliminating 14" Residual Heat Removal (RHR) Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced reports is further identified in Affidavit CAW-03-1606 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by AmerenUE.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-03-1606 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

cc: S. J. Collins
G. Shukla/NRR

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared H. A. Sepp, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

H A Sepp

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 13th day
of March, 2003

Sharon L. Fiori

Notary Public

SHARON L. FIORI
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Manager, Regulatory and Licensing Engineering, in Nuclear Services, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company LLC.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Electric Company LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-15983-P, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant" (Proprietary); WCAP-16019-P, "Technical Justification for Eliminating 10" Accumulator Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant" (Proprietary); and WCAP-16020-P, "Technical Justification for Eliminating 14" Residual Heat Removal (RHR) Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant" (Proprietary) dated February 2003 for Callaway Nuclear Power Plant, being transmitted by AmerenUE letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse Electric Company LLC for Callaway Nuclear Power Plant is expected to be applicable for other licensee submittals

in response to certain NRC requirements for justification of Leak Before Break (LBB) application. The proprietary information was provided by Westinghouse Electric Company LLC.

This information is part of that which will enable Westinghouse to:

(a) Provide documentation of the actual margins relative to flaw size.

(b) Provide the application of the methodology to determine LBB margins.

(c) Assist the customer in obtaining NRC approval by responding to NRC questions.

Further this information has substantial commercial value as follows:

(a) The information reveals the distinguishing aspects of a method prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to process, the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar products and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

COPYRIGHT NOTICE

ATTACHMENT THREE

PROPOSED FSAR CHANGES
(for information only)

Models, typically shown in Figure 3.7(B)-13, were used to perform soil-structure interaction analyses for all three sites. For each site, the site dependent soil properties were used. The vertical dimension of each soil element is equal to or less than Cs/5f, where Cs is the lowest soil element shear wave velocity reached during iterations and f is the highest frequency of interest to be transmitted through the soil profile. The highest frequency used was 25 Hz. In the analyses for the same buildings with site dependent soil parameters, the structural elements remained unchanged.

The site dependent soil properties consisted of strain dependent damping and modulus relationships for each material. In general, the soil properties are nonlinear in character. An iterative process was used to obtain equivalent linear properties which are strain dependent. The methods generally used for such an analysis are included in the computer program FLUSH.

3.7(B).2.5 Development of Floor Response Spectra

Acceleration time-histories obtained from the FLUSH finite element analyses were used in computing the floor response spectra for the major seismic Category I structures. The spectra were generated following the procedures outlined in Section 5.2 of BC-TOP-4-A, using the SPECTRA computer program (see subparagraph 3.8A.12).

3.7(B).2.6 Three Components of Earthquake Motion

Procedures for considering the three components of earthquake motion in determining the seismic response of structures, systems, and components follow the recommendations of Regulatory Guide 1.92 and are described in Section 4.3 of BC-TOP-4-A and Section 5.1 of BP-TOP-1. INSERT A

3.7(B).2.7 Combination of Modal Responses

Combination is done according to the criterion of "the square-root-of-the-sum-of-the-squares" (SRSS).

Section 4.2.1 of BC-TOP-4-A describes the techniques used to combine modal responses for structures and equipment. For piping systems, closely spaced modes were determined per NRC Regulatory Guide 1.92, Equation 4.

3.7(B).2.7.1 Significant Dynamic Response Modes

The static load equivalent or static analysis method involves the multiplication of the total weight of the equipment or component member by the specified seismic acceleration. Multiple degree-of-freedom systems which may have had frequencies in the resonance region of the amplified response spectra curves were analyzed by using a static load of 1.5 times the peak acceleration or the applicable floor response spectra to account for the contribution of higher modes. Multiplication factors less than 1.5 were not used.

INSERT A

The "100-40-40" method from ASCE Standard 4-86, "Seismic Analysis of Safety-Related Nuclear Structures," may be used for combining the orthogonal components of earthquake motion for the design of the 'C' loop steam generator cubicle secondary shield wall.

Design of the primary shield wall is performed, using the strength design methods described in ACI-318.

3.8.3.4.3 Secondary Shield Walls

The secondary shield walls are designed to resist all of the applicable loads, including those due to differential pressure and temperature resulting from a LOCA, RCS component support forces, OBE and SSE, dead and live loads from the operating floor and intermediate platforms and walkways, and those loads resulting from a postulated pipe break. INSERT B

Analysis of the secondary shield walls is performed, using classical techniques and the SAP computer program described in Appendix 3.8A. Design for the effects of postulated pipe breaks is performed using BN-TOP-2.
and ANSYS s

The finite element model used for analyzing the secondary shield walls consists of a three-dimensional model of one-half of the structure in plan about an axis of symmetry. An additional finite element model is used for analyzing these secondary shield walls at the pressurizer. Appropriate boundary conditions are modeled to simulate actual conditions at the axis of symmetry and at the intersections with the base slab, refueling canal walls, floors, and RCS component supports. The analysis for time-dependent loads, such as those for differential pressure and pipe rupture reaction forces, is performed in a manner similar to that used for the primary shield wall. The finite element models used for the secondary shield walls are shown in Figures 3.8-79 through 3.8-82.

Design of the secondary shield walls is performed, using the strength design methods described in ACI-318.

3.8.3.4.4 Refueling Canal Walls

The refueling canal walls are designed to resist all of the applicable loads, including those due to differential pressure and temperature resulting from a LOCA, RCS component support forces, OBE and SSE, hydrostatic loading during the refueling operation, dead and live loads from the operating floor and intermediate platforms and walkways, and those loads resulting from a postulated pipe break.

Analysis of the refueling canal walls is performed, using classical techniques and the SAP computer program described in Appendix 3.8A. Design for the effects of postulated pipe breaks is performed using BN-TOP-2.

The finite element model used for analyzing the refueling canal walls consists of a three-dimensional model of the entire structure. Appropriate boundary conditions are modeled to simulate actual conditions at the intersections with the base slab, secondary shield walls, primary shield wall, floors, and RCS component supports. The analysis for time-dependent loads, such as those for differential pressure and pipe rupture reaction forces, is performed in a manner similar to that used for the primary shield wall. The

INSERT B

The design of the 'C' loop steam generator cubicle secondary shield wall does not have to consider the dynamic effects of postulated pipe breaks in the RCS primary loop and Class 1 branch lines for pipes with a diameter of 10 inches and larger.

c. Extent of Application

The program was used to modify the Bechtel time-history accelerograms to comply with NRC Standard Review Plan Section 3.7.1.

3.8A.1.9 Bechtel CE 798, Engineering Analysis System (ANSYS)

a. Description

ANSYS is a large-scale, general purpose finite element computer program with applications to many classes of engineering problems. Structural analysis methods include static options for the solution of elastic, plastic, and nonlinear large and small deflection problems. Also, dynamic options are available to perform nonlinear transient, harmonic response and mode-frequency analysis. The finite element library is extensive and includes beam, spar, plate, shell, and nonlinear gap elements.

The matrix displacement method of finite element analysis is used in the formulation of the problem, and equations are solved by the wave front method.

b. Validation

The ANSYS program was licensed from Swanson Analysis Systems, Inc. (SASI), which has supplied a complete set of documentation including a user's manual, verification report, and theoretical manual. These documents are on file with Bechtel Data Processing.

c. Extent of Application

The program was used to perform a stress analysis of embedded base plates. INSERT C

3.8A.1.10 Bechtel CE 800, Bechtel Structural Analysis Program (BSAP)

a. Description

The program performs the static and dynamic analyses of linear, elastic, three-dimensional structures, using the finite element method. The finite element library contains truss and beam elements, plane and solid elements, plate and shell elements, axisymmetric (torus) elements, and special boundary (spring) elements.

Element stresses and displacements are solved for either applied loads or temperature distributions. Concentrated loads, pressures, or gravity loads may be applied. Temperature distributions are assigned as an appropriate

INSERT C

and for the access opening in the 'C' loop steam generator cubicle secondary shield wall.

b. For equipment procured as commercial grade and upgraded for safety-related use, an evaluation of equipment design in accordance with utility procedures for upgrading commercial equipment, provided that the program elements of A.1.d above are followed.

REGULATORY GUIDE 1.90 REVISION 1 DATED 8/77

Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons

DISCUSSION:

The recommendations of this regulatory guide are not applicable to the Callaway application, since the containment design does not utilize grouted tendons.

REGULATORY GUIDE 1.91

Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.92 REVISION 1 DATED 2/76

Combining Modal Responses and Spatial Components in Seismic Response Analysis

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Sections 3.7(B).2.6, 3.7(B).2.7, 3.7(B).3.6, 3.7(B).3.7, 3.7(N).2.6, 3.7(N).2.7, 3.7(N).3.6, and 3.7(N).3.7. INSERT D

REGULATORY GUIDE 1.93 REVISION 0 DATED 12/74

Availability of Electric Power Sources

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Technical Specifications.

REGULATORY GUIDE 1.94

Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants

INSERT D

Section 3.7(B).2.6 describes the allowed use of the "100-40-40" method from ASCE Standard 4-86, endorsed in practice by Draft Regulatory Guide DG-1108 (proposed Revision 2 of Regulatory Guide 1.92), for combining the orthogonal components of earthquake motion for the design of the 'C' loop steam generator cubicle secondary shield wall.

Instrumentation for post-accident monitoring is discussed in Section 7.5.

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Basis

Subcompartments within the containment, principally the reactor cavity, the steam generator loop compartments, and the pressurizer compartment, are designed to withstand the transient differential pressures and jet impingement forces of a postulated pipe break. Venting of these chambers maintains the differential pressures within the structural limits. In addition, restraints on the reactor coolant pipes, reactor vessel, steam generators, etc., are designed so that neither pipe whip nor vessel upset forces threaten the integrity of the subcompartments or of the containment structure.

Analysis of the pressure transients in the steam generator compartment and pressurizer compartment has been performed to verify the adequacy of the structural design of these structures under accident conditions. The following is a synopsis of the pipe breaks analyzed:

a. For the steam generator loop compartments, the design basis break is a steam generator inlet elbow longitudinal split with a break flow area of 763 square inches, a double-ended steam generator outlet nozzle break restrained to a break flow area of 436 square inches, and a double-ended reactor coolant pump outlet nozzle break restrained to a break flow area of 236 square inches.

b. The pressurizer compartment is divided into two compartments: 1) the pressurizer vault and 2) the pressurizer surge line compartment.

   The design basis break for these subcompartments is the double-ended pressurizer surge line break. In addition to this break, the pressurizer spray line break and the three break cases from the steam generator loop compartment analysis were considered in the selection of the design analysis break. In all cases, the pressures in the pressurizer compartment were substantially lower than those resulting from the pressurizer surge line break.

INSERT E

6.2.1.2.2 Design Features

All design features provided for alleviating pressure buildup within the subcompartments are discussed in the subcompartment design evaluation in Section 6.2.1.2.3. Reference 2 describes the design features which limit the movement of the pipe after the postulated break.

INSERT E

In accordance with NRC approval of WCAP-10691-P, WCAP-15983-P, WCAP-16019-P, and WCAP-16020-P, all of the above design basis breaks have been excluded from the structural design basis for Callaway Plant. Pipe breaks less than 10-inches in diameter remain as part of the structural design basis.

through 6.2.1-15. The nodalization model for the analyses is given in Figures 6.2.1-43 through 6.2.1-55. Only breaks in loop 4 were analyzed, since this loop has the smallest vent area directly to the remainder of the containment due to the presence of the pressurizer, and thus results in the highest pressures.

To ensure conservative design of the loop compartment walls and the equipment supports, the loads calculated for loop 4 were applied to the other three steam generator loop compartments by appropriate translation and rotation of the force vector axes. INSERT F The volumes of the subcompartments, as well as the initial conditions prior to the transient, are given in Table 6.2.1-22.

As with the reactor cavity analysis, the node boundaries were selected wherever significant restrictions to flow occurred. A sensitivity study was performed in which the number of nodes in the steam generator compartment was varied. The resulting forces on the compartment walls and on the equipment in all cases were less than the forces calculated with the 59-node model. Therefore, it was assumed that the nodalization employed in the original model was both adequate and conservative. All major obstructions, such as columns, pumps, tanks, grating, and the steam generators, were considered in the calculation of the subcompartment volumes and vent areas. In addition, the values for volume were reduced by 5 percent to allow for minor obstructions, such as cable trays, supports, and various piping. The principal obstructions within the steam generator loop compartments were the reactor coolant pumps and the steam generators. Flow through the reactor cavity was neglected. The flow coefficients associated with the flow paths were calculated in the same manner as for the reactor cavity. The head loss coefficients used in the calculation of the flow coefficients, as well as the vent areas and l/a's for each flowpath, are listed in Table 6.2.1-23.

The fluid flow from one subcompartment to another was calculated, using the homogeneous frozen flow option in the analysis. The peak pressures for each subcompartment are listed in Table 6.2.1-22. The complete pressure histories for those subcompartments near the break for each of the three break cases analyzed are shown in Figures 6.2.1-56, 6.2.1-57, 6.2.1-61, and 6.2.1-69. When the subcompartment pressures were applied to their projected areas on the steam generator and the reactor coolant pump, the forces were determined on these pieces of equipment. The forces on the reactor coolant pump and the steam generator are shown in Figures 6.2.1-58, 6.2.1-59, 6.2.1-62 through 6.2.1-67,

INSERT F

The 'C' loop steam generator cubicle secondary shield wall in the area of the access opening has been analyzed with compartment pressure loads specific to that area.