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February 7, 1983

In reply, please
refer to LAC-8850

DOCKET NO. 50-409

Director, Nuclear Reactor Regulation
ATTN: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: DAIRYLAND POWER COOPERATIVE
LA CROSSE BOILING WATER REACTOR (LACBWR)
PROVISIONAL OPERATING LICENSE NO. DPR-45
SEP TOPIC III-6, SEISMIC DESIGN CONSIDERATIONS

REFERENCE: (1) NRC Letter, Crutchfield to Linder, dated November 19, 1982
(2) NRC Letter, Crutchfield to All SEP Owners (Except San Onofre) dated June 17, 1981
(3) Draft Memo, Crutchfield to Eisenhut, dated March 11, 1981
(4) DPC Letter, Linder to Crutchfield, LAC-7181, dated October 14, 1980

Gentlemen:

DPC has reviewed the draft safety evaluation for the seismic design of the La Crosse Boiling Water Reactor, SEP Topic III-6, Seismic Design Considerations and III-11, Component Integrity (Reference 1). The seismic design considerations have been an ongoing consideration since DPC purchased the reactor section of the LACBWR facility and assumed the total operating responsibilities for LACBWR. As early as 1973, DPC embarked on numerous seismic design analysis and studies. Analysis and/or review of topics on soil liquifaction, integrity of structures and piping systems, and electrical equipment anchorage have been addressed. Unfortunately, the criteria for seismic design considerations have been continually evolving and this has complicated application of the older studies to current criteria. An additional complication was a lack of a firm site specific ground spectra acceptable to the Atomic Energy Commission or Nuclear Regulatory Commission. This spectra and return frequency was not assigned and approved until June 1981 (Reference 2). The methodology used in Reference 2 establishes a level of .11g for La Crosse, Big Rock and Palisades and this further states that this value, exceeds the 1,000 year spectra for these plants at frequencies greater than 2 to 3 Hz. The return frequency is assigned at 1,000 years although Reference 2 indicates actual return periods to be closer to 5,000 or 10,000 years. Technical Reference 10 of Reference 4 states that the NRC has

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determined that, *"the return period for an earthquake resulting in a peak acceleration of .12 G would be at least 1,000 years and that the actual return period could be an order of magnitude higher. The LACBWR site is located in the Central Stable Region where historically the seismic activity is very low. Using seismicity data developed by the TERA Corporation for Lawrance Livermore Laboratory and the NRC, in conjunction with a computer program designed to perform seismic risk analysis, Dames & Moore has determined that the return period for an earthquake of this size is at least 10,000 years and more likely between 10,000 and 100,000 years."* This assignment by consensus of more conservative spectra should be considered in the final determination of the scope of required seismic review and work at LACBWR.

DPC proposes to assess the consequences of failure of those structures, systems and components required to mitigate core melt accidents, both during and after a postulated seismic event, for either the consequences of failure or the upgrading required to resist seismic events:

(1) Safe Shutdown of the reactor; and

(2) The integrity of the reactor coolant primary system pressure boundary; and

(3) The integrity of selected fluid and electrical distribution systems related to safe shutdown and engineered safety features.

Attachment 1 addresses some of the specific concerns raised by the Safety Evaluation Report/Technical Evaluation Report that DPC is providing additional information or clarification.

Attachment 2 summarizes the effort DPC has expended in good faith since 1973 while the AEC and NRC was developing its criteria for assigning specific site spectra. This summary does not, however, reference or detail the extreme magnitude of the effort and expense DPC has committed to demonstrate that the phenomenon of liquification does not occur at the LACBWR site.

DPC also provides in Attachment 3 an outline of the actions it proposes to accomplish the remaining seismic review.

DPC maintains that any assessments and resulting modifications over and above those proposed do not provide any significant additional benefit to the health and safety of the public. This view is supported by a NRC draft paper (3/11/81) based on a report prepared by the Sandia National Laboratories, New Mexico, of their study, entitled "Licensing Actions and Technical Review Efforts for Smaller and Older Licensed Power Reactors", (Reference 3). LACBWR is one of the facilities referred to in the decision paper. The basis of the referenced decision paper's conclusion that LACBWR should be exempted from many of NRC's regulations is that even in the case of the worst possible accident, the actual risk to the public is minimal. The referenced decision paper states:

> *"Using very conservative assumptions of a core melt resulting in very large isotopic releases, containment failure and varied evacuation schemes (or relocation back into the intercepted area) the results show that there are no acute fatalities for the four plants studied. The importance of this preliminary information lies in the realization that very conservative assumptions were used in predicting the insult to the population.*
>
> *Further, the results do not disclose a need for concern, even given these assumptions. They point out rather vividly that decisions by the NRC staff regarding implementation of additional measures at these plants could and should be tempered with the "real world" realization that, because of low power level and low area population, the impact of any realistic postulated event will be minimal."*

Not only are the risks to the public from a serious accident minimal, the referenced decision paper found that facilities like LACBWR are uniformly operated safely:

> *"What is immediately obvious about these facilities is their relatively low (≤ 200 MWe) power level, their relative isolation from population centers and their age (between 10 and 20 years of commercial operation). None of these facilities has ever had an accident or incident determined to be detrimental to the health and safety of the public, although each has at times had heightened public concern regarding the safety of nuclear power... The overall good safety record, the small size of the plants and their distance from population centers should by themselves be cause for consideration of amended licensing policies toward the facilities, but additional information developed later in this paper adds significant weight to this idea."*

Based on this NRC decision paper it is clear that the safety consequences of modification to the LACBWR facility are minimal. Dairyland is not seeking to minimize its responsibility for safe reactor operation. Although LACBWR has been classified as a Power Reactor, its inventory of nuclear material and, consequently, the risk to the public and employees from any accident is low. LACBWR is the smallest commercial power reactor in the country. As a 50 MWe plant, LACBWR's nuclear material inventory is many times smaller than other commercial power reactors.

It is DPC's desire to continue to operate the La Crosse Boiling Water Reactor in a safe and cost effective manner for the benefit of the public. The proposed seismic review plan is an attempt to meet the intent of the NRC regulation and still maintain a reliable and economical generating facility.

Very truly yours,

DAIRYLAND POWER COOPERATIVE

Frank Linder

Frank Linder
General Manager

FL:RMB:eme

cc: J. G. Keppler, Regional Administrator, Region III
NRC Resident Inspector

Enclosures

ATTACHMENT 1

CONTAINMENT BUILDING

NRC CONCERN:

Clarify the stress and capacity calculations for the lower columns of the inner shield structure. Specifically, resolve the discrepancy between calculations on Pages B-208 and B-222 of Reference 2 of Reference 1.

DPC RESPONSE:

In the calculation on Page B-208 the working strength method was used with the assumption that only the main column portion of the cross-section takes the loading. Also, the reduction in moment induced tensile stress due to dead load compression was neglected. This conservative approximation did not yield acceptable results. Consequently the calculation was repeated on Page B-222 with the above conservatisms eliminated. The results were then acceptable.

NRC CONCERN:

Evaluate the stability of the containment against overturning along the interfaces between the internal concrete structure and the steel containment, between the steel containment and the basemat, and between the basemat and the pile group.

DPC RESPONSE:

The critical interfaces for stability are between the steel containment and the basemat and between the basemat and the pile group. The attached calculation sheets (Refer to Enclosure 7) show that the factor of safety against tipping for the steel containment at the steel containment basemat interface is 1.79 for the SSE. The factor of safety against overturning along the interface between the basemat and pile group is greater than 1.79 due to the additional basemat weight and the broader base. This conclusion is based on the assumption that the piles have sufficient compression bearing capacity which will be discussed later.

NRC CONCERN:

The ultimate pile capacity of 400 kips is four times the rated capacity. Clarify the methodology that was used to derive this number.

DPC RESPONSE:

The piles were driven to a resistance of six (6) blows per inch for the final twelve (12) to eighteen (18) blows using a McKiernan-Terry double acting hammer having a rated energy, E, of 16,000 ft-lbs per blow. The rated static pile capacity of 100 kips was derived from the well-known Engineering News formula which gives:

$$R = \frac{2E}{S+0.1} = \frac{2(16,000)}{1/6 + 0.1} = 120 \text{ kips}$$

This formula provides an "allowable" static load not the ultimate bearing resistance. It has a built-in theoretical factor of safety of six (6) with respect to the ultimate dynamic bearing capacity. However, based on comparisons with lead tests, a factor of six is thought to be too high. Reference 5 indicates the safety factor is thought to be between two (2) and five (5) providing an ultimate load capacity of from 240 to 600 kips. Reference 6 indicates that in the range of 5 to 7 blows per inch, the safety factor may range upward from 3 which would give an ultimate load capacity greater than 360 kips. Based on the above, a value of 400 kips was selected as a reasonable ultimate dynamic load capacity for these piles.

NRC CONCERN:

Evaluate the sloshing effects in the overhead water storage tank on the steel containment.

DPC RESPONSE:

The sloshing effect of the water in the overhead storage tank was evaluated in the analysis performed by Gulf United Services in 1974, (Reference 4, Section 4.3.1). The moments developed by the horizontal water sloshing is negligible in the comparison to the available moment in the soil. The overhead storage tank water sloshing effects are negligible and thus were neglected in the more recent evaluations.

NRC CONCERN:

Evaluate the connection between the piles in tension and the basemat. What is the maximum number of piles under tension at any one time.

DPC RESPONSE:

A re-examination of the tensile capability of the connection between the pile and the basemat indicates that it would be more appropriate to assume that significant tensile loads could not be transferred to the piles. Consequently, the maximum compressive pile load was recalculated assuming zero tension capability. For this case, the maximum compressive pile load was calculated to be approximately 230 kips compared to the 192 kips previously reported. This load is still less than the estimated 400 kips ultimate dynamic bearing load capability for the piles. This verifies the containment building stability discussed earlier.

NRC CONCERN:

The licensee should perform re-evaluation analysis for the following additional safety related equipment or provide justification for not evaluating.

(1) Reactor Vessel and Reactor Vessel Support

DPC RESPONSE:

Refer to the analysis performed by Gulf United Services in 1974, Reference 4 Section 4.3.2. The stress levels in the reactor vessel and internal support structure during the specified seismic event were found to be negligibly small.

TURBINE BUILDING

NRC CONCERN:

Clarify if the basemat is considered as a lumped mass in the seismic analysis models. Justify if it is not.

DPC RESPONSE:

The basemat was not included as a lumped mass in the seismic analysis model of the Turbine Building. Due to the base with respect to the building, neglecting the basemat will have only minimal effect on the basic natural frequency calculated for the building. Thus, the effect on in-structure response spectra will also be minimal. However, the weight of the basemat will impose a compressive force on the piles due to dead load and downward seismic response. The compressive forces on the piles are much smaller than the ultimate capacity taken to be 400 kips. Hence, inclusion of the additional loading from the basemat will not change the conclusion regarding their adequacy.

LACBWR STACK AND GENOA 3 STACK

NRC CONCERN:

Both the LACBWR stack and Genoa 3 Stack were found to be overstressed.

DPC RESPONSE:

The analysis of the 500 foot Genoa 3 stack concludes that the SSE will cause a stack failure 150 to 200 feet from the top. A stack failure at this location would not endanger the LACBWR facility. Refer to Reference 8, Figure 8.1, "Seismic and Structural Analysis of the Genoa 3 Stack using the NRC Site-Specific Ground Response Spectra."

The analysis of the 350 foot LACBWR stack concludes that the SSE will not cause a stack failure, refer to Enclosure 9, Figure 8.1, analysis performed by Nuclear Energy Services, Inc., entitled, "Seismic and Structural Analysis of the LACBWR Stack Using the NRC Site-Specific Ground Response Spectra."

Neither the Genoa 3 stack or the LACBWR stack will present a hazard or threaten a reactor safe shutdown after a seismic event.

GENERAL CONCERNS

NRC CONCERN:

It is not known if all computer codes have been officially verified. From the review of the reports submitted during the meeting held in NES office on August 10 and 11, 1982, it seems that STARDYNE is the only computer code used. STARDYNE is a public domain program. Licensee's not required to verify and to document this program. Clarification is needed if there are other in-house computer programs used.

DPC RESPONSE:

In addition to STARDYNE, the ANSYS, PIPESD, and WERCO computer codes were also used. These are also well-known public domain programs. No in-house codes were used in the evaluation.

NRC CONCERN:

Verify the adequacy of all structural connections. Wherever the ductile behavior is relied upon for structural integrity, the connection should have the capacity to allow the development of ultimate member strength.

DPC RESPONSE:

No reliance was placed on ductile behavior for structural integrity.

NRC CONCERN:

The load combinations used to perform the seismic re-evaluation analysis of the HPCS piping systems component supports were not provided. The licensee should provide these to NRC so they can be reviewed for compliance with SEP requirements.

DPC RESPONSE:

The load combination used to perform the seismic re-evaluation analysis of the HPCS piping systems component supports is given below.

Dead Load + Thermal $\pm$ Seismic Anchor Mov. $\pm$ Dynamic

The seismic analysis of the HPCS piping was performed using a two dimensional spectra where the vertical earthquake component was combined with each of the two horizontal components resulting in two dynamic load cases. For the purposes of component support design, the reaction forces due to the two dynamic load cases were combined by SRSS techniques in the following manner.

Dynamic Reaction Forces = $[(x + y)^2$ eq. $+ (y + z)^2$ eq.$]^{1/2}$.

The calculation of the HPCS reaction forces is more conservative than that outlined in Reg. Guide 1.92.

NRC CONCERN:

Information on electrical system anchorage has not been supplied by licensee.

DPC RESPONSE:

Enclosed as Enclosures 1, 2, 3, 4 and 5 is a summary and supporting documentation showing the extensive work performed on electrical equipment anchorage. Due to the number of copies of prints required and the short response time, DPC will include the title block on modification drawings and not full prints.

ATTACHMENT 2

SEISMIC ANALYSIS OF LACBWR SYSTEMS AND STRUCTURES

The enclosed attachment is a summary of the analyses which have been performed to date on seismic design SEP Topic III-6.

1. Seismic and Structural Analysis for the La Crosse Boiling Water Reactor Shutdown Condenser", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, November 1981, Document No. 81A0044, Revision 1.

   Technical analysis report was forwarded to the NRC in Reference 9.

   SUMMARY OF ANALYSIS -

   This report, prepared for Dairyland Power Cooperative (DPC) presents the results of the structural analysis performed by Nuclear Energy Services, Inc. (NES) to verify the adequacy of the shutdown condenser during the Safe Shutdown Earthquake (SSE). The dead load, live load, pressure, and seismic stresses were found using conventional approximate methods. The main concern comes from local stresses placed on the shutdown condenser during a seismic event.

   There are two areas where these local stresses exist, at the saddle supports and at the nozzles. After evaluating the primary stresses on the shutdown condenser, local stresses were evaluated. An in-depth study was made to find local bending stresses generated by the saddle supports during the seismic event. Nozzle loads were also checked with respect to local effects.

   This report presents the final results for all parts of the shutdown condenser with the exception of the interals (tube bundle, baffles, tube bundle supports, etc.) and the 14 inch vent line nozzle.

   SUMMARY OF CONCLUSIONS -

   The results of this analysis indicate that the major structural components of the shutdown condenser will maintain their integrity during the SSE event.

   Preliminary analysis of the internals, using simplified analytical methods, indicates that stresses in the tube bundle may exceed their allowable during the SSE event. Further investigation will be required before a conclusion can be made on the internals.

   The local stresses generated by the 14-inch diameter vent line will depend on the final support configuration on that line. These stresses will be maintained below allowables by suitable support modifications to the vent line, if required. The 14-inch vent line is the subject of a separate report (Item 13 of this attachment).

   The validity of the results presented in this report is contingent upon completion of the modifications recommended in item 2 of this attachment.

2. "Seismic and Structural Analysis for the La Crosse Boiling Water Reactor Shutdown Condenser Platform:, prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, January 1982, Document No. 81A0045.

   Technical analysis report was forwarded to the NRC in Reference 9.

   SUMMARY OF ANALYSIS -

   This report, prepared for Dairyland Power Cooperative, presents the results of seismic and stress analyses of the Shutdown Condenser Platform for the LACBWR Nuclear Power Station. The seismic and stress analyses are performed in accordance with the requirements of US NRC Standard Review Plan, Section 3.8.3.

   SUMMARY OF CONCLUSIONS -

   Based on the results of this analysis, it was concluded that the existing platform would require additional members and supports. With these additional members, stresses due to seismic, dead weight and live load, are within the design requirements.

3. "Seismic and Structural Analysis of the LACBWR Stack Using the NRC Specific Ground Response Spectra", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, September 1982, Document No. 81A0046, Revision 1.

   Technical analysis report Revision 1 is enclosed (Enclosure 9).

   SUMMARY OF ANALYSIS -

   This report, prepared for Dairyland Power Cooperative (DPC), presents the results of the seismic/structural analysis of the LACBWR stack using the NRC site-specific ground response spectra for the Safe Shutdown Earthquake Event (SSE).

   Linear seismic analysis, using the site specific spectra and modal superposition, was performed to determine the response of the LACBWR stack for the SSE Event. Soil structure interaction effects were included using the information provided by Dames & Moore. The foundation springs reflect the updated information of the most recent boring program. The seismic response of the stack was compared to the load carrying capacities of the stack at corresponding elevations.

   SUMMARY OF CONCLUSIONS -

   From the results of the analysis, it has been concluded that the LACBWR stack is capable of withstanding the specified safe shutdown Earthquake. Shear forces, moments in all nodes of the stack model were found to be less than their ultimate capacities.

4. "Seismic and Structural Analysis for the La Crosse Boiling Water Reactor 1-B Diesel Generator Building", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, January 1982, Document No. 81A0047.

   Technical analysis report was forwarded to the NRC in Reference 9.

   SUMMARY OF ANALYSIS -

   This report, prepared for Dairyland Power Cooperative (DPC) presents the results of the seismic/structural analysis of the 1-B Diesel Generator Building using the NRC site-specific ground response spectra for the Safe Shutdown Earthquake (SSE) Event. It became necessary to verify the structural adequacy of the building under the Systematic Evaluation Program (SEP). The 1-B Diesel Generator Building contains safety related equipment.

   A static analysis using a coefficient of 1.5 times the peak acceleration of the NRC site-specific ground response spectra applied to the lumped masses at the column nodes was used to evaluate the building.

   SUMMARY OF CONCLUSIONS -

   It was concluded from the results of the analysis that the 1-B Diesel Generator Building is capable of withstanding a SSE seismic event. Forces and stresses in all structural elements of the building were determined. All member stresses were found to be less than their allowable values. Evaluation of the effects of the SSE on the non-structural concrete block walls is not included as part of this analysis.

5. "Seismic and Stress Analysis of LACBWR Feedwater Piping System", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, June 1975, Document No. 81A0087.

   Technical analysis report was forwarded to the NRC in Reference 10.

   SUMMARY OF ANALYSIS -

   This report, prepared for Dairyland Power Cooperative, presents the results of seismic and stress analyses of the feedwater piping system for the LACBWR Nuclear Power Station. The seismic and stress analyses are performed in accordance with the design requirements for Class 2 piping components of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components", 1974.

   SUMMARY OF CONCLUSIONS -

   This study shows that by providing adequate seismic restraints (snubbers) at critical locations of the feedwater system, the stresses in the piping due to a seismic event have been reduced to acceptable values. It is concluded that the stresses due to seismic, deadweight, pressure and thermal expansion loadings, combined according to the ASME Code rules for Class 2 components, satisfy the design requirements given in the Code.

6. "Seismic and Stress Analysis of LACBWR Main Steam Piping System", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, August 1975, Document No. 81A0088.

   Technical analysis report was forwarded to the NRC in Reference 10.

   SUMMARY OF ANALYSIS -

   This report, prepared for Dairyland Power Cooperative, presents the results of seismic and stress analyses of the main steam piping system for the LACBWR Nuclear Power Station. The seismic and stress analyses are performed in accordance with the design requirements for Class 2 piping components of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components", 1974.

   SUMMARY OF CONCLUSIONS -

   This study shows that by providing adequate seismic restraints (snubbers) at critical locations of the main steam system, the stresses in the piping due to a seismic event have been reduced to acceptable values. It is concluded that the stresses due to seismic, dead weight, pressure and thermal expansion loadings, combined according to the ASME Code rules for Class 2 components, satisfy the design requirements given in the Code.

7. "Seismic and Stress Analysis of LACBWR Recirculation Piping System", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, November 1975, Document No. 81A0089.

   Technical analysis report was forwarded to the NRC in Reference 10.

   SUMMARY OF ANALYSIS -

   This report, prepared for Dairyland Power Cooperative, presents the results of seismic and stress analysis of the recirculation piping system for the LACBWR Nuclear Power Station. The seismic and stress analyses have been performed in accordance with the design requirements for Class 2 piping components of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components", 1974.

   SUMMARY OF CONCLUSIONS -

   This study shows that by providing adequate seismic restraints (snubbers) at critical locations of the recirculation system, the stresses in the piping due to a seismic event have been reduced to acceptable values. That is, the stresses due to seismic, dead weight, pressure and thermal expansion loadings, combined according to the ASME Code rules for Class 2 components, would satisfy design requirements given in the Code with the addition of seismic restraints.

8. "Seismic and Analysis of the LACBWR High Pressure Core Spray Suction Line Piping System", preapred by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, July 1976, Document No. 81A0090.

   Technical analysis report was forwarded to the NRC in Reference 10.

   SUMMARY OF ANALYSIS -

   This report, prepared for Dairyland Power Cooperative, presents the results of seismic and stress analyses of the High Pressure Core Spray (HPCS) piping system suction line for the LACBWR Nuclear Power Station. The seismic and stress analyses are performed in accordance with the design requirements for Class 1 piping components of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components", 1974.

   SUMMARY OF CONCLUSIONS -

   This study shows that by providing seismic restraints (snubbers) at critical locations of the HPCS suction line, the stresses in the piping due to a seismic event can be reduced to acceptable values. That is, the stresses due to seismic, dead weight, pressure and thermal expansion loadings, combined according to the ASME Code rules for Class 1 components would satisfy the design requirements given in the Code with the addition of seismic restraints.

9. "Seismic and Stress Analysis of the LACBWR High Pressure Core Spray Discharge Line Piping System", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, June 1982, Document No. 81A0091, Revision 1.

   Technical analysis report Addendum 1 is enclosed (Enclosure 10).

   SUMMARY OF ANALYSIS -

   This report, prepared for Dairyland Power Cooperative, presents the results of seismic and stress analyses of the High Pressure Core Spray (HPCS) piping system discharge line for the LACBWR Nuclear Power Station. The seismic and stress analyses are performed in accordance with the design requirements for Class 1 piping components of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components", 1974.

   SUMMARY OF CONCLUSIONS -

   This study shows that by providing seismic restraints at critical locations of the HPCS discharge line, the stresses in the piping due to a seismic event can be reduced to acceptable values. That is, the stresses due to seismic, deadweight, pressure and thermal expansion loadings, combined according to the ASME Code rules for Class 1 components would satisfy the design requirements given in the Code with the addition of seismic restraints. The detail stress analysis also indicates 2900 permissible HPCS operating cycles.

10. "Seismic and Structural Analysis of the Genoa 3 Stack Using the NRC Site-Specific Ground Response Spectra", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, November 1980, Document No. 81A0040.

    Technical analysis report was forwarded to the NRC in Reference 8.

    SUMMARY OF ANALYSIS -

    This report, prepared for Dairyland Power Cooperative (DPC), presents the results of the seismic/structural analysis of the Genoa 3 stack using the NRC site-specific ground response spectra for the Safe Shutdown Earthquake Event (SSE).

    Linear seismic analysis, using the site specific spectra and modal superposition, was used to determine the response of the Genoa 3 stack for the SSE Event. Soil structure interaction effects were included using the information provided by Dames & Moore. The foundation springs reflect hte updated information of the most recent boring program. The seismic response of the stack is compared to the load carrying capacities of the stack at corresponding elevations.

    SUMMARY OF CONCLUSIONS -

    From the results of the analysis, it has been concluded that under an SSE seismic event, the Genoa 3 stack will experience a failure 150 to 200 feet from its top. The surviving 300 to 350 feet of the stack will remain upright and attached to its foundation mat. Since the Genoa 3 stack is located approximately 400 feet from the LACBWR Reactor Containment Building and other safety related structures, the failed section of the stack should not impact on these structures.

11. "Seismic Evaluation of the La Crosse Boiling Water Reactor", prepared by Gulf United Services for Dairyland Power Cooeprative, January 1974, Document No. SS-162.

    Technical analysis report was forwarded to the NRC in Reference 4.

    SUMMARY OF ANALYSIS -

    This report presents the results of the Gulf United geotechnical investigation of the surrounding region, and the site of the La Crosse Boiling Water Reactor. Seismic analysis of the following structures was also conducted:

    1. Reactor Containment Building
    2. LACBWR Stack
    3. Genoa 3 Stack
    4. Waste Disposal Building
    5. Turbine Building

SUMMARY OF CONCLUSIONS -

Through the evolution of the Systematic Evaluation Program, Topic III-6, many guideline changes have occrred. Because of these changes additional studies have been performed to supplement the original Gulf United studies.

12. "Seismic and Structural Analysis for the La Crosse Boiling Water Reactor Turbine Building", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, July 1982, Document No. 81A0048.

Technical analysis report was forwarded to the NRC in Reference 9.

SUMMARY OF ANALYSIS -

This report, prepared for Dairyland Power Cooperative (DPC), presents the results of the seismic/structural analysis of the turbine building. The NRC's site specific spectra was used in this analysis for the Safe Shutdown Earthquake Event (SSE).

SUMMARY OF CONCLUSIONS -

Most structural elements of the turbine building including the reinforced concrete lower portion, the turbine foundation, and the piles foundation were evaluated for the SSE event and were found acceptable. The analysis indicated certain members will be overstressed during the SSE event. The top portion of the building is a basic structural steel-framed building with bracing members in the walls to resist lateral loads. This bracing and some of the diagonal roof bracing were calculated to be overstressed.

STRUCTURE MODIFICATION -

The 1A Diesel Generator has been anchored to prevent displacement during a seismic event.

13. "Seismic and Stress Analysis of the LACBWR 14" Shutdown Condenser Vent Piping System", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, Document No. 81A0051.

Technical analysis report was forwarded to the NRC in Reference 9.

SUMMARY OF ANALYSIS -

This report, prepared for Dairyland Power Cooperative, presents the results of seismic stress analysis of the 14" vent line of the shutdown condenser system. The shutdown condenser is a safety-related system at LACBWR. The analysis performed is in accordance with the design requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division I, Class 2 piping code of 1980. The original design of LACBWR's piping was to the ANSI B-31.1 piping code. Class 2 piping analysis methods are considered to be appropriate and conservative.

SUMMARY OF CONCLUSIONS -

With a small modification to a vertical support to resist lateral loads the analysis indicate that the stresses are low compared to the allowable stresses. The bellows expansion joint included in this line undergoes an applied torque, which is considered an undesirable form of loading for expansion joints per Section III of the ASME 1980 code. The bellows applied torques have been calculated to be 5034 lb-in and 8009 lb-in for OBE and SSE events respectively. However, an evaluation based on the bellows manufacturers recommendations shows that the bellows can withstand the applied torque. The maximum allowable applied torque is 94,000 lb-in.

14. "Seismic and Structural Analysis for the LACBWR Containment Building", prepared by Nuclear Energy Services for Dairyland Power Cooperative, July 1982, Document No. 81A0049.

Technical analysis report was forwarded to the NRC in Reference 7.

SUMMARY OF ANALYSIS -

This report, prepared for Dairyland Power Cooperative (DPC), presents the results of the seismic/structural analysis of the LACBWR containment building using the NRC site-specific ground response spectra for the Safe Shutdown Earthquake event (SSE).

Linear seismic analysis, using the site specific spectra and modal superposition methods, was performed to determine the response of the LACBWR containment building for the SSE event. Soil structure interaction effects were included using the information provided by Dames & Moore. The foundation springs reflect the updated information from the most recent boring program. The combination of deadload and seismic stress of the containment building is compared to the allowable stress.

SUMMARY OF CONCLUSIONS -

From the results of the analysis, it has been concluded that the containment building is capable of withstanding the specified Safe Shutdown Earthquake. Forces, moments and stresses in all structural elements of the building were determined. All member stresses were found to be less than their allowable values.

15. "Seismic and Stress Analysis of the LACBWR Manual Depressurization System", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, November 1982, Document No. 81A0052.

Technical analysis report is enclosed (Enclosure 8).

SUMMARY OF ANALYSIS -

This report, prepared for Dairyland Power Cooperative (DPC), presents the results of the seismic/stress analysis of the Manual Depressurization Piping System, (MDS).

The seismic/stress analysis was performed in accordance with the design requirements for Class 2 piping components of the ASME Boiler and Pressure Vessel Code, Seciton III, Division 1, "Nuclear Power Plant Components", 1980.

SUMMARY OF CONCLUSIONS -

A seismic analysis of the MDS piping system with its existing support configuration indicated that the stresses due to the SSE earthquake would be much greater than the allowable stress values. The analysis demonstrates that the stresses in the piping due to a seismic event can be reduced to acceptable values by providing adequate seismic restraints at three critical locations and modifying one existing support of the Manual Depressurization System. Preliminary details for the required new supports and modifications were provided. In addition to the new required supports and modifications it is recommended that three existing supports be disabled.

16. "Safety-Related Electrical Equipment Anchorage for the La Crosse Boiling Water Reactor", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, January, 1983, Document No. 81A0041, Revision 3.

Technical analysis report is enclosed (Enclosures 2 and 3).

SUMMARY OF ANALYSIS

The safety related electrical equipment anchorage was evaluated by Nuclear Energy Services, Inc. for Dairyland Power Cooperative.

SUMMARY OF CONCLUSIONS

The following equipment was found to need seismic anchorage:

(1) Generator Battery Rack

(2) 480 Volt Essential Bus

(3) Reactor Battery Rack

(4) Battery Charger

(5) 1A Diesel Starting Battery Rack

(6) 1B Diesel Generator Control Panel

(7) 1B Diesel Generator Building Electrical Equipment

(8) Reactor Water Level Transmitter #1 and #2

(9) Auxiliary Distribution Panel for 125 Volt DC Generator Plant Bus

(10) Electrical Equipment Room Modification

Additional details including anchorage designs and stress calculations can be found in Enclosures 2 and 3.

17. Analysis was performed by Nuclear Energy Services, Inc. for Dairyland Power Cooperative pertaining to emergency power equipment anchorage. The items which were found to need no modifications are listed below:

    Turbine Building 120-V Bus. Aux. Dist. Panel
    Solatron for Turbine Bldg. Reg. Bus.
    Transformer for Turbine Bldg. 120-V Bus 480-240/120-V
    1-B Emergency Diesel Generator
    1-B Diesel Fuel Oil Day Tank
    1-B Emergency Diesel Gen. Starting Batteries and Racks
    1-B 480-V Ess. Bus.
    1-B Static Inverter
    1-B Diesel Building Battery Charger
    ACS Valve DC Motor Starter
    ACS Flow Transmitter
    Reactor Water Level No. 3 Transmitter
    HPCS Motor 1A
    HPCS Motor 1B
    1C Static Inverter
    1A 1 KVA Inverter
    1A Inverter Input and Output Breakers
    1A Inverter Meter Panel
    120-V AC Non-Int. Bus. 1A
    120-V AC Non-Int. Bus. 1A Fuse Panel

    Refer to Enclosure 6 for additional information on the analysis methods used.

18. A summary of the safety related electrical equipment appears in Enclosure 1. The safety related electrical equipment is listed in tabular form. A brief description of the equipment location, support structure, evaluations performed, and if modification were made is summarized in these tables.

19. Additional Tests Performed on Safety Related Electrical Equipment -

    From the original Gulf United 1974 Report equivalent seismic reaction load were calculated from the SSE accelerations at various elevations within the containment building. Using these reaction forces applied at the component centroid, a pull test was performed on various safety related electrical equipment to determine if the component would remain intact during a seismic event. Refer to Enclosure 4 for additional details on the tests run.

20. High Pressure Service Water Piping and Alternate Core Spray Piping Study -

    A seismic analysis of the High Pressure Service Water Piping System and Alternate Core Spray Piping System is presently being conducted by Nuclear Energy Services, Inc. Result of this analysis should be available in early 1983.

ATTACHMENT 3

The following information outlines the additional steps which will be taken by DPC to assess the as-designed structures and systems to improve the capability of LACBWR to resist an earthquake. The scope of the systems and structures discussed in this attachment are the minimum necessary to provide safe shutdown. A discussion of these systems and structures follows.

CONTAINMENT BUILDING STABILITY

Containment Building Stability is required to minimize the effect on the public in the event of this postulated seismic event which could lead to a LOCA. The seismic analysis of the LACBWR Containment Building performed by Nuclear Energy Services, Inc. (References 7 and 9) show this structure to be adequate to withstand the SSE.

CONTROL ROD DRIVE MECHANISMS

In order to ensure that the reactor can be safety shut down following a seismic event, the deflection of the control rod drive mechanism must be evaluated to ensure insertion of the control rods. Simplified, analysis will be performed and modifications added if required to ensure control rod insertion.

HIGH PRESSURE CORE SPRAY

DPC has completed the seismic analysis of the High Pressure Core Spray System. A total of 31 restraints have been added to the system to date and one remaining restraint will be installed in the near future. This will complete the work on this piping to provide integrity of the high pressure injection system during seismic events.

DPC PROPOSES

DPC proposes to evaluate the following systems required for safe shutdown in one or more of the methods listed below:

(1) Perform a study to determine the consequence of system failures and assess the potential impact on the health and safety of the public if such systems are not modified to resist seismic events. Determine an effective heat removal method which would be upgraded to provide for the reactor safe shutdown, or;

(2) Perform analysis on the system to determine the as-built capability to withstand a Safe Shutdown Earthquake (S.S.E.), or;

(3) Perform a system walk down and using a piping handbook method, restrain the system using good engineering judgement and verify thermal margins are not unacceptably effected by the additional restraints, or;

(4) For those systems previously analyzed to older criteria either install restraints as indicated in the studies or reanalyze with the intention of relocating some restraints and possibly eliminating others.

The systems requiring further evaluations of at least a portion of their piping to the above criteria (if methods 2-4 are utilized) include;

(1) Feedwater (A)

(2) Main Steam (A)

(3) Manual Depressurization (A)

(4) Forced Circulation (A)

(5) Boron Injection

(6) Decay Heat

(7) Shutdown Condenser (A)

(8) Turbine Building (A)

(9) Emergency Service Water Supply (B)

(10) High Pressure Service Water/Alternate Core Spray (B)

(11) Associated Electrical Controls and Instrumentation (A)

Analysis states for above system;

(A) Analysis Performed

(B) Analysis in Progress

REFERENCES

1. NRC Letter, Crutchfield to Linder, dated November 19, 1982

2. NRC Letter, Crutchfield to All SEP Owneers (Except San Onofre), dated June 17, 1981

3. Draft Memo, Crutchfield to Eisenhut, dated March 11, 1981

4. DPC Letter, J. Madgett to A. Giambusso, LAC-2788, dated October 9, 1974

5. Lambe, T. W., and R. V. Whitman (1969), Soil Mechanics, New York, Wiley Page 506

6. Havers, J., and F. Stubbs (1971), Handbook of Heavy Construction, New York McGraw Hill, Pages 27-46, 27-47

7. "Seismic and Structura' Analysis for the LACBWR Containment Buiding," prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, July 1982, This report was forwarded in Reference 9.

8. DPC Letter, Frank Linder to Dennis M. Crutchfield, LAC-7255, dated November 26, 1980

9. DPC Letter, Frank Linder to Dennis M. Crutchfield, LAC-84[illegible], dated July 22, 1982

10. DPC Letter, Frank Linder to Director NRR, LAC-5478, dated September 27, 1978

ENCLOSURES

(1) Summary of Safety-Related Electrical Equipment

(2) Safety-Related Electrical Equipment Anchorage, Document No. 81A0041, Revision 3

(3) Supporting Calculations for Item 2

(4) Static Pull Test Summary Sheets

(5) Attachment 2 of DPC Letter, Linder to Crutchfield, LAC-7484, dated April 23, 1981

(6) NES Transmittal, 5101-803, dated June 2, 1982

(7) NES Transmittal, 5101-832, dated January 17, 1983

(8) "Seismic and Stress Analysis of the LACBWR Manual Depressurization System", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, November 1982, Document No. 81A0052, Revision 0

(9) "Seismic and Structural Analysis of the LACBWR Stack Using the NRC Site-Specific Ground Response Spectra", prepared by Nuclear Energy Service, Inc. for Dairyland Power Cooperative, September 1982, Document No. 81A0046, Revision 1

(10) "Seismic and Stress Analysis of the LACBWR High Pressure Core Spray Discharge Line Piping System", prepared by Nuclear Energy Services, Inc. for Dairyland Power Cooperative, May 1977, Addendum 1, June 1982, Document No. 81A0091.

(1)

SUMMARY OF ELECTRICAL EQUIPMENT

KEY TO ATTACHMENT

EXAMPLES OF TYPE SUPPORT

1. Bolted to Equipment
2. Bolted to Concrete Wall
3. Bolted to Concrete Slab
4. Bolted to Block Wall
5. Welded to Embedded Channel
6. Slide Drawer with stops
7. Strapped

REFERENCE OR STATUS

A. To Be Evaluated

B. Evaluation in Progress

C. Attachment No. 2, LAC-7484

D. To Be Modified

E. Modification Commenced

F. Intentionally Blank

G. NES Task-063

H. Test Data or Documented Review on File at Site

I. To be Replaced as Part of Environmental Qualification Program

J. Task Placed on Hold Status 12/81

K. Temporarily Secured

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIP. I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Benchboard "E"		Control Room	Turbine Bldg. Control Room 668'	3,5		Nuclear Instr. Indicators Log Count Ch. 1 43-38-804	1	H, C				J
						Period Ch. 1 42-38-805	1	H, C				
						Log Count Ch. 2 42-38-806	1	H, C				
						Period Ch. 2 42-38-807	1	H, C				
						Log N Ch. 3 42-38-808	1	H, C				
						Period Ch. 3 42-38-809	1	H, C				
						Log N Ch. 4 42-38-810	1	H, C				
						Period Ch. 4 42-38-811	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Benchboard "E" (Cont'd)						Nuclear Instr. Indicators	1	H, C				
						Power Level Ch. 5 42-38-812	1	H, C				
						Power Level Ch. 6 42-38-813	1	H, C				
						Power Level Ch. 7 42-38-814	1	H, C				
						Power Level Ch. 8 42-38-815	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Panel G		Control Room	Turbine Bldg. Control Room 668'	3,5		Nuclear Instr. Ch. 7 42-38-507	1	H, C				J
						Nuclear Instr. Ch. 8 42-38-508	1	H, C				
						Scaler Source Range Monitor Ch. 1 & Ch. 2 42-38-509	1	H, C				
						Radiation Monitoring Recorder 45-43-801	1	H, C				
						Containment Radiation Montitors 73-43-501 73-43-502 73-43-503	6	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Panel F		Control Room	Turbine Bldg. Control Room 668'			Nuclear Instr. AGS Channels (Rear Panel F)	1	H, C				J
						Containment Vessel Level Power Supply 37-42-401	1	H, C				
						ATWS Relay 50-42-602AT	1	H, C				
						ATWS Relay 50-42-604AT	1	H, C				
						ATWS Relay 50-42-601AT	1	H, C				
						ATWS Relay 50-42-712AT	1	H, C				
						ATWS Relay 50-42-605AT	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Panel E		Control Room	Turbine Bldg. Control Room 668'	3,5		Nuclear Instr. Dual Pen Rec. Power Range Ch. 7 & 8 42-38-803	1	H, C				J
						Reactor Water Level #2 Power Supply 50-42-401	1	H, C				
						Reactor Water Level #2 Remote Amplifier 50-42-303	1	H, C				
						ATWS Relay 50-42-603AT	1	H, C				
						Reactor Water Level Remote Amplifier 50-42-302	1	H, C				
						Power to Flow Square Root Converter 2A 50-37-503	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Panel E (Cont'd)						Power to Flow Square Root Converter 2B 50-37-504	1	H, C				
						Power to Flow Square Root Converter 1A 50-37-501	1	H, C				
						Power to Flow Square Root Converter 1B 50-37-502	1	H, C				
						Nuclear Instr. Dual Pen Rec. Ch. 3 & 4 42-38-801	1	H, C				
						Nuclear Instr. Dual Pen Rec. Wide Range Ch. 5 & 6 42-38-802	1	H, C				
						Reactor Flow Recorder 50-37-801	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Panel E (Cont'd)						Reactor Water Level #3 Power Supply 50-42-403	1	H, C				
						Reactor Water Level #3 Level Indicator 50-42-811	1	H, C				
						Reactor Water Level #1 P/S 47-85-406 (Dist. Panel)	1	H, C				
						Reactor Water Level #2 Indicator 50-42-802	1	H, C				
						Reactor Water Level #1 Rec. 50-42-801	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Safety Panel	Panel D	Reactor Protection	Turbine Bldg. Control Room 668'	3,5		Rx Water Level #1 Safety Drawer 50-42-501	1	H, C				J
						Rx Water Level #2 Safety Drawer 50-42-502	1	H, C				
						Rx Pressure #1 Safety Drawer 63-35-501	1	H, C				
						Rx Pressure #2 Safety Drawer 63-35-302	1	H, C				
						P/F #1 Safety Drawer 50-37-505	1	H, C				
						P/F #2 Safety Drawer 50-37-506	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Panel D		Control Room	Turbine Bldg. Control Room 668'	3,5		Alternate Core Spray Low Flow Alarm 38-37-601	1	H, C				J
						ACS Flow Indicator 38-37-801	1	H, C				
						ACS AC Valve Motor Control Switch	1	H, C				
						ACS DC Valve Motor Control Switch	1	H, C				
						Containment Vessel Internal Pressure Indicator 37-35-815	1	H, C				
						Containment Vessel Internal Pressure Indicator 37-35-810	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Panel D (Cont'd)						Containment Vessel Internal Pressure Power Supply 37-35-401	1	H, C				
						Containment Vessel Liquid Level Selector Switch 37-31-701	1	H, C				
						Containment Vessel Liquid Level Indicator 37-43-801	1	H, C				
Reactor Plant Battery Charger	74-92-002	Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	3	Yes	Breakers, Relay Meters	1	H, C				SK-5101-063 -7 H
Reactor Plant Batteries	74-91-001	Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	Sitting On Battery Rack	No							SK-5101-063 -6 H

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Reactor Plant Battery Rack		Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	3	Yes							SK-5101-063 -6 H

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Generator Plant Battery Charger		Essential Power	Turbine Bldg. Electrical Equipment Room 654'	3	Yes	Input & Output Breakers	1	H, C				SK-5101-063 -1 H
Generator Plant Batteries		Essential Power	Turbine Bldg. Electrical Equipment Room 654'	Sitting On Battery Racks	No							SK-5101-063 -2 H
Generator Plant Battery Racks		Essential Power	Turbine Bldg. Electrical Equipment Room 654'	None	Yes							SK-5101-063 -2 H

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Reactor Relay Cabinet		Reactor Protection	Turbine Bldg. Electrical Equipment Room 654'	3	Yes	Relays	1	H, C				SK-5101-063 -1 H
Turbine Bldg. MCC 1A		Essential Power	Turbine Bldg. Electrical Equipment Room 654'	3	Yes	Rx Plant Battery Charger Bkr.	1	H, C	Ladder to BBD			SK-5101-063 -1 H
						480-120 Volt Breaker	1	H, C				
						G.P. Main Steam Shutoff Valve Breaker 64-11-003	1	H, C				
						Control Room Emergency Lighting Bkr.	1	H, C				
						ACS AC M.O. Valve Breaker 38-30-001	1	H, C				
						Turbine Bldg. 120-Volt Dist. Panel	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Turbine Bldg. MCC 1A (Cont'd)						Turbine Bldg. 120-Volt Reg Dist. Panel	1	H, C				
Turbine Bldg. 120-Volt Bus Aux. Dist. Panel		Essential Power	Turbine Bldg. Electrical Equipment Room 654'	1	No							H NES 5101-803

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
1B Non-Interruptible Bus Breakers Panel		Essential Power	Electrical Equipment Room El. 654'	1	Yes	Tie to 1B Static Inverter Breaker	1	H, C				SK-5101-063 -1 H
						Main Control BBD "E"	1	H, C				
						Main Control BD. D Breaker W/D 41-503704	1	H, C				
						Radiation Monitor Panel G3 Breaker	1	H, C				
						Safety System Panel D2 (41-503764) (41-503801) Breaker	1	H, C				
						Nuclear Instr. Panel G1 (41-503806) Breaker	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Rx Plant 125-Volt DC Bus	78-87-001	Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	1	Yes	Main Control BBD "D" Breaker 41-503698 41-503701	1	H, C				SK-5101-063 -1 H
						480-Volt Ess. Switchgear Feed Breaker 41-503666	1	H, C				
						ACS DC Valve Breaker 41-503828 41-503775	1	H, C				
						Station Under Voltage Relays Breaker 41-503634	1	H, C				
						Inverter 1A Non-Interruptible Bus Breaker 41-503677	1	H, C				
						MSIV & Bypass	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Reactor Plant 125-Volt DC Bus (Cont'd)						Reactor Plant Battery to Reactor Plant Bus	1	H, C				
						Reactor Plant Charger to Reactor Plant Battery	1	H, C				
1C Static Inverter		Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	2	No							H NES-5101-803
Undervoltage Relay Cabinet		Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	3	Yes	Undervoltage Relays for 480-Volt Ess. 1A & 1B	1	H, C				SK-5101-063 -1
						480-Volt Turbine Bldg. MCC 1A	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Generator Plant 125-Volt DC MCC		Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	3	Yes	Aux. Dist. Panel Breaker	1	H C				SK-5101-063 -1 H
						Control Power #1 Breaker	1	H, C				
						Control Power #2 Breaker	1	H, C				
						BBD "D" Breaker	1	H, C				
						3 Breakers for C.B. Control Power	1	H, C				
						Relay Panel	1	H, C				
						Gen. Battery to Gen. Plant 125-Volt DC Bus	1	H, C				
						Gen. Battery Charger to Gen. 125-Volt DC Bus	1	H, C				
						Gen. Battery Charger to Gen. Battery	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Aux. Dist. Panel for 125-Volt DC Gen. Plant Bus		Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	4	Yes	1C Static Inverter Bkr.	1	H, C				SK-5101-063-13
						Gen. Plant Batteries Breaker	1	H, C				
						Gen. Plant Battery Charger & Associated Breakers	1	H, C				
						Diesel Bldg. Protective Relays	1	H, C				
Solatron For Turbine Bldg. Reg Bus	74-81-001	Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	3	No							H 5101-803
Transformer For Turbine Building 120-Volt Bus 480-240/120 Volt	78-80-001	Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	2	No							H 5101-803

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
1A 1 KVA Inverter		Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	2	No							H 5101-803
1A Inverter Input & Output Breakers		Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	2	No							H 5101-803
1A Inverter Meter Panel		Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	2	No							H 5101-803
120-VAC Non-Interruptible Bus 1A		Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	2	No							H, C 5101-803
120-VAC Non-Interruptible Bus 1A Fuse Panel		Essential Power	Turbine Bldg. Electrical Equipment Room El. 654'	2	No							H, C 5101-803

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
1B Emergency Diesel Generator	78-83-002	On-Site Emergency Power	1B Diesel Bldg. El. 641'	3	No				Fuel Oil Tank	3	Yes 5101-803 D	H 5101-803
1B Diesel Generator Control Panel	78-83-001	On-Site Emergency Power	1B Diesel Bldg. El. 641' D.G. Room	5	Yes	Relays & Controls	1	H, C				SK-5101-063-9
1B Emergency Diesel Gen. Starting Batteries & Racks	78-83-902	On-Site Emergency Power	1B Diesel Generator Bldg. El. 641'	3	No							H 5101-803

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
480-Volt Ess. Bus 1B		On-Site Emergency Power	1B Diesel Generator Bldg. El. 641'	5	No	1B D.G. Output Breaker (74-79-044) 452 EGB	1	H, C				H SK 5101-063-10 5101-803
						Inter Bus Tie 452 TBB Bkr. 74-77-043	1	H, C				
						Emergency Core Spray Pump 1B 452 ECCB Bkr. (74-77-013)	1	H, C				
						480-Volt Diesel Bldg. 1B Feed MCC 452 DB Breaker 74-77-045	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
480-Volt Diesel Bldg. 1B MCC		On-Site Emergency Power	1B Diesel Generator Bldg. El. 641'	5	Yes	Diesel Bldg. Battery Charger Bkr. 74-32-061	1	H, C				SK 5101-063-10
						Diesel Engine Radiator Fan Breaker 78-32-004	1	H, C				
						Diesel Bldg. Uninterruptible Supply 74-32-062	1	H, C				
						Diesel Bldg. Distribution Transformer 74-32-060	1	H, C				
						Distribution Panel 74-80-004	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	HAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
1B Static Inverter		Essential Power	1B Diesel Generator Bldg. El. 641'	3,5	No	Transformer Relay & Controls	1	H, C				H 5101-803
1B Diesel Building Standby Batteries		Essential Power	1B Diesel Generator Bldg. El. 641'	3	No							D SK 5101-063-11
1B Diesel Building Battery Charger		Essential Power	1B Diesel Generator Bldg. El. 641'	3,5	No	Transformer Relays & Controls	1	H, C				H, B 5101-803
125-VDC Diesel Bldg. Main Dist. Bus		On-Site Emergency Power	1B Diesel Generator Bldg. El. 641'	5	Yes	1B Static Inverter Bkr. 74-32-067	1	H, C				SK 5101-063-10
						DC Control Power Ess. Switchgear 1B Breaker 74-32-063	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
125 VDC Diesel Bldg. Main Dist. Bus (Cont'd)						D.G. Control Panel Feed Breaker 74-32-068	1	H, C				
						D.G. Bldg. Engine Gen. Control Crt. 74-32-064	1	H, C				
						Diesel Bldg. Main Battery Feed 74-32-070	1	H, C				
						Diesel Bldg. Main Charger Feed 74-32-066	1	H, C				
						Diesel Bldg. Battery/ Charger Tie 74-32-065	1	H, C				

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
480-Volt Ess. Bus 1A		Essential Power	Turbine Bldg. Penetration Room El. 640'	3	Yes	Feed Breaker 452 EGA 78-79-001	1	H, C				SK-5101-063 -4
						1A Emergency Core Spray Pump Breaker 452 ECC A 74-77-012	1	H, C				
						Turbine Bldg. MCC 1A Feed Breaker 74-77-011	1	H, C				
						Inter Bus Tie Breaker 452 TBA	1	H, C				
ACS Valve Motor	38-30-001 (AC)	Alternate Core Spray	Turbine Bldg. Mezz. Level 654'	3								B
ACS Valve Motor	38-30-002 (DC)	Alternate Core Spray	Turbine Bldg. Mezz. Level 654'	3								B

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
ACS Valve DC Motor Starter	For 38-30-002	Alternate Core Spray	Turbine Bldg. Mezz. Level 654'	2	No							5101-803
1A Emergency Diesel Generator	78-83-001	On-Site Emergency Power	Turbine Bldg. El. 640'	3	Yes	Output Breaker 78-31-701	1	C	Fuel Oil Day Tank	4	B	Facility Change 78-81-3
						Battery Charger	1	C				
1A Diesel Generator Starting Battery	78-83-901	On-Site Emergency Power	Turbine Bldg. El. 640'	Strapped to Rack	Yes							SK-5101-063-8
1A Diesel Generator Starting Battery Rack		On-Site Emergency Power	Turbine Bldg. El. 640'	2	Yes							SK 5101-063-8
Containment Bldg. Electrical Penetration		Containment	Containment Wall Penetration 640'	Welded to Shell	No							81A0049

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Offgas Vent Header Solenoid Valve	55-25-014	Containment Isolation	Turbine Bldg. Pipe Tunnel 635'		No							H
Containment Bldg. Pressure Switches	37-35-701 37-35-702 37-35-703	Containment Building	Turbine Bldg. Penetration Room 640'	1 1 1	No No No							H H H
ACS Flow Transmitter	38-37-301	Alternate Core Spray	Turbine Bldg. Mezz. 654'	2	No							H 5101-803 5101-814
Containment Vessel Internal Pressure Transmitter	37-35-301	Containment Vessel Monitoring	Turbine Bldg. Pipe Tunnel 629'	3	No							H
Containment Vessel Internal Pressure Transmitter	37-35-302	Containment Vessel Monitoring	Turbine Bldg. Pipe Tunnel 629'	3	No							H

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Containment Vessel Internal Level Transmitter	37-42-301	Containment Vessel Monitoring	Turbine Bldg. Pipe Tunnel 629'	3	No							H
Containment Vessel Internal Level Transmitter	37-42-302	Containment Vessel Monitoring	Turbine Bldg. Pipe Tunnel 629'	3	No							H
Containment Ventilation Solenoid Valve	73-25-016	Containment Isolation	Containment 643'	1	No							H
Containment Ventilation Solenoid Valve	73-25-017	Containment Isolation	Containment 643'	1	No							H
Containment Ventilation Solenoid Valve	73-25-018	Containment Isolation	Containment 643'	1	No							H

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Containment Ventilation Solenoid Valve	73-25-019	Containment Isolation	Containment 643'	1	No							H
Containment Ventilation Solenoid Valve	73-25-003	Containment Isolation	Containment 643'	1	No							H
Containment Ventilation Solenoid Valve	73-25-008	Containment Isolation	Containment 643'	1	No							H
P/F 1A Transmitter	50-37-301	Reactor Protection	Containment 633'	2	No							H
P/F 1B Transmitter	50-37-304	Reactor Protection	Containment 633'	2	No							H
P/F 2A Transmitter	50-37-302	Reactor Protection	Containment 633'	2	No							H
P/F 2B Transmitter	50-37-303	Reactor Protection	Containment 633'	2	No							H

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
MISV Bypass Solenoid Valve	64-25-002	Containment Isolation	Containment 633'	1	No							H
MSIV Bypass Solenoid Valve	64-25-003	Containment Isolation	Containment 633'	1	No							H
Solenoid for Decay Heat Isolation Valve 56-25-001	56-25-002	Containment Isolation	Containment 633'	1	No							H
Reactor Offgas Solenoid	55-25-013	Containment Isolation	Grade Floor Containment 643'	2	No							H
Reactor Offgas Solenoid	55-22-022	Containment Isolation	Grade Floor Containment 643'	2	No							H
Reactor Water Level #1 Transmitter	50-42-302	Reactor Protection	Containment 659'	2	No							SK-5101-063-12

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Reactor Water Level #3 Transmitter	50-42-306	Reactor Protection	Containment 659'	2	No							I 5101-803
Reactor Water Level #2 Transmitter	50-42-303	Reactor Protection	Containment 659'	2	No							SK-5101-063-12
Radiation Monitor	73-43-201	Containment Isolation	Containment 667'	None	Yes							
Radiation Monitor	73-43-202	Containment Isolation	Containment 667'	None	Yes							A, K
Radiation Monitor	73-43-203	Containment Isolation	Containment 667'	None	Yes							
MSIV Solenoid	61-22-005	Hydraulic Valve Accumulator	Containment Grade 643'	1	No							H, I

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
HPCS Motor (1A)	53-06-001	High Pressure Core Spray	Containment 667'	3	No							5101-803
HPCS Motor (1B)	53-06-002	High Pressure Core Spray	Containment 667'	3	No							5101-803
LPCS Inlet to Reactor Solenoid Valve	53-25-005	Low Pressure Core Spray	Containment Approx. 672'	2	No							H
MDS Solenoid	62-25-015	Manual Depressuri-zation	Containment Approx. 714'	1	No							H
MDS Solenoid	62-25-016	Manual Depressuri-zation	Containment Approx. 714'	1	No							H
Nuclear Instr. Detectors, Cables, & Junction Boxes Ch. 1 thru 8		Reactor Protection	Containment	1,2,7	No							NA

SUMMARY OF INVESTIGATION OF ANCHORAGE AND SUPPORT OF
SAFETY RELATED ELECTRICAL EQUIPMENT AND ITEMS THAT MAY DAMAGE THIS EQUIPMENT

EQUIPMENT NAME	EQUIPMENT I.D.	SYSTEM IN WHICH INSTALLED	LOCATION BLDG. & ELEV.	TYPE OF SUPPORT	WAS ANCHORAGE MODIFIED SINCE 1/1/80	INTERNALLY ATTACHED COMPONENTS			ITEMS THAT COULD POTENTIALLY INTERACT WITH THIS EQUIPMENT			I.D. OF DOCUMENT SUPPORTING CONCLUSION
						EQUIPMENT NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	NAME & I.D.	TYPE OF SUPPORT	WAS SUPPORT EVAL.'D	
Containment Ventilation Solenoid Valve	73-25-004	Containment Isolation	Containment 643'	1	No							H
Containment Ventilation Solenoid Valve	73-25-007	Containment Isolation	Containment 643'	1	No							H
Decay Heat Blowdown Solendoid	56-25-002	Containment Isolation	Containment 633'	1	No							H
Containment Post-Accident Sample Valves		Post-Accid Sample	Containment	2	No							H
		Post-Accid. Sample	Electrical Penetration	1	No							H
		Post-Accid. Sample	Electrical Penetration	1	No							H
		Post-Accid. Sample	Containment	2	No							H